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**Los Alamos Scientific Laboratory Waste
Management Technology Development Activities
Summary Progress Report 1979**

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INVESTIGATION OF VOLCANIC TUFF AS A MEDIUM FOR DISPOSAL OF LOW LEVEL RADIOACTIVE WASTE

Waste Disposal Site at Los Alamos

*Horizontal and Vertical
Core Drilling at
LASL Disposal Sites*

*Volcanic Rocks (in orange)
in the western U.S.*

Waste disposal pits and shafts at Los Alamos are excavated in volcanic tuff. The radionuclide retention and hydrologic properties of tuff are being determined through field studies at existing disposal sites. Volcanic rocks, including tuffs, are widespread in the western U.S. and may prove to be a versatile medium for both near surface burial and alternative methods of low level waste disposal.

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**LOS ALAMOS SCIENTIFIC LABORATORY
WASTE MANAGEMENT TECHNOLOGY DEVELOPMENT ACTIVITIES**

**Summary Progress Report
1979**

Compiled by

L. J. Johnson

ABSTRACT

Summary reports on the Department of Energy's Nuclear Energy-sponsored waste management technology development projects at the Los Alamos Scientific Laboratory describe progress for calendar year 1979. Activities in airborne, low-level, and transuranic waste management areas are discussed. Work progress on waste assay, treatment, disposal, and environmental monitoring is reviewed.

I. INTRODUCTION

Waste management technology development activities sponsored by the Department of Energy (DOE) Division of Waste Products, Office of Nuclear Waste Management, at the Los Alamos Scientific Laboratory (LASL) cover a broad range of technical tasks and disciplinary expertise. Studies on airborne waste filtration systems response to pressure transients and collection efficiency are contributing to improved air cleaning systems. Low-level waste studies on disposal site selection, monitoring, alternatives, and environmental engineering are aimed at improved understanding and operation of solid radioactively contaminated waste disposal facilities. The transuranic waste tasks fall primarily into the areas of waste assay instrumentation development and waste treatment engineering studies. The summary statements in this progress report provide an insight into the objectives and general progress for each of these areas. Detailed reports in each study area are developed as program progress allows. Recent publications and presentations are listed at the end of each section.

Whereas the emphasis of this waste management development and demonstration work is aimed at airborne, low-level or transuranic waste needs, the results are applicable to other waste types. Industrial application of these efforts is being pursued. For example, the waste measurement instrumentation and the controlled-air incinerator design and specification information is being provided to several commercial concerns.

A related work activity recently developed overview data on waste production rates in the United States.¹ These data bring a useful perspective on the relative quantities of municipal, industrial, radioactive, mining and milling, agricultural, and utility wastes being generated. It can be seen from this information that there is potential technology need for resource recovery, reuse, and disposal. Indications are that radioactive waste operational practices and techniques are relatively advanced when compared to other types of waste such as hazardous or toxic wastes. It is likely that DOE-developed waste technology will find non-nuclear applications.

REFERENCE

1. D. M. Lacombe, "An Overview of Solid Waste Generation in the United States," Los Alamos Scientific Laboratory report LA-8712-MS (December 1979).

II. AIRBORNE WASTE MANAGEMENT STUDIES

A. Investigation of Pressure Transients in Nuclear Filtration Systems

[W. S. Gregory, H. L. Horak, and P. R. Smith]

The objective of this investigation is to evaluate the response of nuclear facility air filtration system components to explosively driven or tornado-induced pressure transients. The project results will be particularly applicable to nuclear plants, but can also be applied to other facilities such as synthetic fuel plants. The program is primarily experimental and we are constructing a shock tube to simulate explosive transients and using an existing blowdown facility to simulate tornado transients.

Initially, we will be concerned with subjecting high-capacity, high-efficiency particulate air (HEPA) filters to shock and overpressure transients and in later phases we will examine response of other ventilation system components. If the new higher capacity filters can withstand tornado and explosion transients, the higher air-flow capacity and longer life will result in a reduction of volume and disposal cost of airborne waste. If these devices are found to respond unsatisfactorily, the long-range goal of the program will be to produce recommendations for improved design of filtration devices.

In the past reporting period (Jan.-Dec. 1978) we have been primarily concerned with construction of a large 36-in.-diam shock tube that will be used to simulate explosion transients. The shock tube is essentially complete and preliminary firings have been performed. Testing of high-capacity HEPA filters for simulated tornado transients has been performed and structural limit results from some European HEPA filters have been obtained.

The shock tube is shown in Fig. 1 and is 914 mm in diameter with a 12.1-m driver section and a 38.1-m driven section. A double-diaphragm technique is used to control driver-firing pressure. This method allows us to reduce diaphragm costs by eliminating the need for machine-scored diaphragms. Figure 2 shows a filter mounted in the end of the shock tube. The shock tube conceptual design and small-scale experiments are reported in Refs. 1 and 2.

We intend to control the total impulse that is imposed on the test specimen. That is, we will control both peak pressure and duration (that is, dwell time) of the high pressure behind the shock wave. A wide range of dwell times can result from internal explosions. Diverse systems within facilities and their geometrical configurations are responsible for part of the variability in dwell times, but other conditions may be even more influential. These conditions result from the character of the material causing the explosions. Fuel cycle operations typically involve gases, vapors, and dust or fine granular material. These materials often have explosive potential and

*Fig. 1.
Shock tube facility.*

*Fig. 2.
Shock tube with filter mounted for testing.*

vary widely in their deflagration or detonation characteristics. We have concluded that it is impossible to pick a single representative dwell time for a detonation wave. Thus we have devised a method to allow variable dwell times with the shock tube.

The method used is suggested by the physical phenomena occurring in the shock tube. Gas at different pressures is separated by a diaphragm. When the diaphragm is ruptured, a compression wave is sent down the low-pressure region of the shock tube while an expansion wave travels in the opposite direction into the high-pressure section. This latter wave is reflected and races back down the shock tube, tending to overtake the shock wave. By varying the length of the driver section of the shock tube, we can obtain any dwell time desired. The dwell time will be the difference between the arrival times of the compression and expansion waves at the test specimen.

We plan to evaluate the effect of total impulses on ventilation system components during the remainder of FY 1980. Initial tests are in progress using standard HEPA filters.

Structural limits for four different types of high-temperature German filters under simulated tornado conditions have been obtained. These results are listed in Table I. A blowdown device was used to establish a pressure differential, at a selected pressurization rate, across the filters. A high-speed movie camera and all instrumentation recorders were started simultaneously.

TABLE I
BREAK PRESSURE OF KARLSRUHE
FILTERS

Filter Test	A (kPa)	B (kPa)	C (kPa)	D (kPa)
1	12.1	10.1	4.2	4.5
2	12.8	10.2	5.9	3.7
3	11.1	10.6	5.4	4.0
4	12.1	10.4	6.2	4.7
5		10.7	6.2	
6		10.9		
Average	12.0	10.5	5.6	4.2

Pressure differential across the filters and dynamic pressure were measured by electronic pressure transducers and recorded by strip chart recorders. A timing mark was simultaneously recorded on the high-speed film and pressure recorder at 10-ms intervals. The actual times of filter failure were then found by observing the high-speed movies of the downstream filter face. The results in Table I are relatively consistent for the repeated tests. This was because in most cases the entire filter pack was discharged from its frame. Filters A and B were from the same manufacturer, with A having a wood frame and B a metal frame. Filter C was a v-type, metal-frame filter of European design. Filter D was made by an American company for the German government. Tests of other European and high-capacity HEPA filters to simulated tornado overpressures are in progress.

REFERENCES

1. W. S. Gregory and P. R. Smith, "Ventilation System Pressure Transients, Proposed Experiments and Shock Tube Conceptual Design," Los Alamos Scientific Laboratory report LA-7413-MS (September 1978).
2. D. LaPlante, P. R. Smith, and W. S. Gregory, "Ventilation System Pressure Transients—Small Scale Shock Tube Results," Los Alamos Scientific Laboratory report LA-7726-MS (April 1979).

PRESENTATION

W. S. Gregory, J. W. Bolstad, H. L. Horak, R. W. Andrae, R. A. Martin, P. K. Tang, P. R. Smith, and C. J. Ricketts, "Analytical and Experimental Investigation of Nuclear Facility Ventilation Systems for Accident Conditions," presented at CSNI Restricted Specialist Meeting on Behavior of Off-Gas and Ventilation Systems Under Accident Conditions, Karlsruhe, F. R. of Germany, October 24-26, 1979.

B. Standardization of HEPA Filter Test Method

[John Elder, Marvin Tillery, and Harry Ettinger]

Standardization of the multiple HEPA filter test method by using the laser single-particle spectrometer¹ is now under development as a limited consensus standard through the DOE Nuclear Standards Office at Oak Ridge National Laboratory (ORNL). The method was proposed as an improved in-place filter test method similar to the present American National Standards Institute (ANSI) N510. This new method will provide greater sensitivity, allow testing of two filter stages at a time, and measure filter efficiency as a function of particle size within the limits of the test aerosol size spectrum. The sensitivity of the method also permits testing very large, single-stage HEPA systems. As part of the earlier development program,² all required auxiliary equipment (a portable dilution system and a high-capacity gas-thermal aerosol generator) was developed and a field demonstration program was initiated. The standard writing and approval process is continuing into FY 1980.

Preliminary drafts of the proposed standard were prepared and reviewed internally by the LASL Industrial Hygiene Group while initial contacts were made with appropriate national consensus organizations such as ANSI, American Society of Mechanical Engineers (ASME), and American Society for Testing Materials (ASTM). These organizations declined developing the method as a national consensus standard primarily because the method is too new to have gained

acceptance in a large number of facilities and does not yet have a large data base to support its applicability. The DOE Nuclear Standards Office accepted the proposal to standardize the method on a limited consensus basis as a suitable vehicle for promptly disseminating the method to DOE facilities now having an established need. This would permit DOE facilities to quantify the performance of their air-cleaning filtration systems.

The limited consensus process has involved writing of the standard by a 5-member writing group and subsequent approval of the standard by a 10- to 12-member review committee, whose members are selected to bring broad experience and knowledge of filter test methods into the standard preparation process. The draft standard was prepared by the writing group during a 3-day meeting at LASL. The method was demonstrated on a 1000-cfm dual-stage system and the preliminary draft was composed from the earlier draft prepared by the Industrial Hygiene Group. This draft was revised by the writing group and submitted to the review committee. The review committee will comment in depth and return the draft to the writing group for reconciliation of comments and submittal to the Office of Nuclear Waste Management (ONWM) for final approval and issue.

REFERENCES

1. B. Schuster and R. Knollenberg, *Applied Optics* 11, No. 7, 1515-1520 (July 1972).
2. American National Standards Institute, "Testing of Nuclear Air-Cleaning Systems," ANSI N5.3-1975.
3. B. Schuster and D. Osetek, "A New Method for In-Place Testing of Tandem HEPA Filter Installations," *AIHA J.* 40, 979-985 (1979).

PRESENTATION

J. Elder, Presentation to the American Society for Testing Materials F 21.20 Subcommittee on Air Filtration, Danvers, Mass., October 1979.

C. Fluorescent Aerosol Test Method Development

(T. Kyle)

In testing air-cleaning systems, there is often ambiguity because background aerosols mask the test aerosol being injected into the system.¹ There is a need for a technique to count and size aerosol particles automatically and identify them as being the test aerosol, and not part of the background aerosol. Single-particle laser spectrometers automatically count and size the aerosols, but have no identification capability. The system developed to overcome this problem has an additional optical detector for identifying the test aerosol on the basis of its specific fluorescent properties.

The fluorescent particle spectrometer was developed as a cooperative effort of LASL and Particle Measuring Systems (PMS), Boulder, Colorado. By following a cooperative development scheme rather than developing a prototype first, the instrument was developed and a commercially manufactured unit is available. This significantly reduced the time necessary for a new

filter testing technology to become available and saved considerable costs. This instrument's performance characteristics are based on specifications developed at LASL to satisfy the requirements for in-place filter testing. Final testing and calibration of the instrument is planned for January 1980 as a joint LASL-PMS effort.

If visible light is used for excitation, particles containing a particular fluorescent dye can be uniquely identified. The fluorescent particle spectrometer contains two optical detectors, one for the scattered light, and a second that blocks light of the excitation wavelength, and sees only the fluorescence induced in the particle. The first detector sizes the particle and, provided the second detector indicates the particle is fluorescent, the particle is counted. If the particle is not fluorescent, no count is permitted, which results in only the test aerosol being counted. In order to allow the instrument to be used for other purposes (not involving fluorescent aerosols) a switch is provided to override the requirement that the particle be fluorescent. This allows the instrument to count all particles, whether fluorescent or not.

The fluorescent particle spectrometer has two size ranges with 16 channels in each range. The large-size range produces the size spectrum of particles from 1 to 3.1 μm as well as the total number of particles in this size range. The small size range provides the same functions, but for particles from 0.1 to 1.7 μm . An automatic paper printout is available on the front of the control panel along with a cathode ray tube which provides a real time display of the size spectrum. The latter is useful when particles are being counted for long periods of time as it allows the operator to monitor the evolving spectrum. The light source is a helium-cadmium laser having a wavelength of 441 nm. The laser beam is confined to the instrument and thus provides no safety hazard in operation.

The instrument was designed for testing multiple-stage HEPA filters in a single test. The combined test will result in a considerable saving of time and improved accuracy. The specific particle identification capability makes it quite desirable for other studies such as tracer work. For example, if it is desired to determine if air from one hood in a laboratory is recirculating back to an air intake, one can inject fluorescent particles in the first hood and measure the particles in the air intake with the fluorescent particle spectrometer. It is possible to visualize similar outdoor situations where it is desirable to measure an identifiable aerosol. The commercial availability will allow the fluorescent particle spectrometer to be used in diverse technological areas.

There are two types of fluorescent particles that will be used with the spectrometer. One is a commercially available thermoplastic solid particle with the fluorescent dye contained in the plastic. The other particles consist of the fluorescent dye dissolved in dioctyl phthalate (DOP). Techniques for producing acceptable aerosols from each of these substances have been determined.

Because initial tests involving thermal generation (which first evaporates and then condenses the liquid DOP), indicate that the aerosol produced is only weakly fluorescent, an alternate technique using a sonic nozzle has been developed. Through the use of compressed air directed into a cavity, a sonic standing wave is created that breaks up large drops entering the cavity.² The unit can produce the narrow size distributions and high concentrations of DOP in the size range needed for HEPA filter testing. The solid particles can also be generated by feeding a suspension of the particles to the sonic nozzle.

The solid particles have the ability to retain their identity after being collected, so it is practical to collect them on a membrane filter, then measure and count the particles on the filter at a later time. Only the generating and sampling equipment need to be taken to the test site, providing for better quality control of the test data. Only a limited quantity of the solid aerosol can be used in the testing process without loading the filter and significantly increasing the pressure drop. A technique for sampling on a membrane filter and then removing the particles

from the filter for automatic sizing has been developed. It appears that a series of two HEPA filters in a standard system can be tested by using only a small quantity of solid particles. This is possible because of the larger sampling rate that can be used when the sample is collected on membrane filters.

The introduction of these methods for testing filters will provide for more cost-effective testing of air-cleaning systems. It will also quantitate performance of the air-cleaning system in terms of particle size and number concentration, thereby providing a realistic assessment of the level of control provided.

REFERENCES

1. B. Schuster and D. Osetek, "A New Method for In-Place Testing of Tandem HEPA Filter Installations," *Am. Ind. Hyg. Assoc. J.* **40**, 979-985 (1979).
2. A. J. Muir, "Sonic Atomizers," *Plant Engineering*, December 28, 1972.

PUBLICATIONS

T. G. Kyle, "Determining Particle Concentration by Statistics," *J. Aerosol Science* **10**, 87-93 (1979).

B. Schuster and T. Kyle, "Pollution Plume Transport and Diffusion Studies Using Fluorescent Lidar," accepted by *Applied Optics*.

III. LOW-LEVEL WASTE MANAGEMENT STUDIES

A. Soil Moisture Studies at LASL Waste Disposal Sites

[W. V. Abeele]

Migration of soil moisture (with the possibility of transport of radionuclides), is strongly dependent on the moisture content of the tuff. Soil-moisture distribution readings are taken monthly (six monitoring holes at Area G and four holes at Area C) with a neutron moisture probe at depth intervals of 0.5 m. Soil moisture is then computed by applying the correct calibration formula. These formulae have recently been revised, incorporating the influence of access hole parameters on neutron moisture probe readings.¹ The revision also took into account the possible interference of soil density on moisture probe readings.

The nuclear density probe was calibrated to determine the dependence of the count ratio on hole diameter at six different densities, obtained by mixing two extremely miscible materials in six differing ratios. The holes were either uncased or cased with polyvinyl chloride (PVC), aluminum, or steel. Extremely high inverse linear correlations existed between hole diameter and count ratios provided the hole was uncased or cased with PVC or aluminum. In these cases, the impact of casing material was insignificant and the count ratio of a hole with an untested diameter could be predicted for a given soil density. By using the count ratio as an independent variable and the soil density as the dependent variable, with hole diameter held constant, a linear regression emerged that enabled prediction of soil density with a high degree of accuracy.

Moisture and density readings were taken in the existing monitoring holes. After subtracting the computed moisture content by volume from the computed (wet) density, a regression analysis was run on 460 readings from Area C using the obtained (dry) density as the independent variable and the moisture ratio by volume as the dependent variable. The dry density was taken as the independent variable because a correlation would logically exist between wet density and moisture (moisture would increase wet density). The regression analysis showed that 1.5% of the variation in moisture content was associated with dry density in Bandelier tuff.

The equation shows a slight direct relationship between moisture and dry density. The relationships between dry density and moisture ratio by volume are shown in Figs. 1 through 4.

The computed density shows no significant difference at the 0.99 confidence level between CP and GP7 holes. The first letter attached to the designation of a hole is the disposal area letter (G, C, ...). The following letter and number indicate the nearness to a specific pit (P1) or shaft (S50). An additional number is used if more than one monitoring hole has been drilled in the vicinity. Readings at GP1-1 and GP1-2 were taken in the pit cover and an expected lower density was found. Low readings in solid tuff from the GS50 holes could be the result of using a smaller hole diameter in the density calibration formula than actually exists. A hole diameter could exceed its nominal value because of vibration during drilling or postdrilling sloughing.

There is no certainty that the same phenomenon is not manifesting itself, in a more subtle way, in other monitoring holes. The computed densities should consequently be considered as minimum values.

REFERENCE

1. W. V. Abeele, "The Influence of Access Hole Parameters on Neutron Moisture Probe Readings," Los Alamos Scientific Laboratory report LA-8094-MS (October 1979).

Fig. 1.
Dry density and moisture profiles at CP2-1 in June 1979.

Fig. 2.
Dry density and moisture profiles at CP5-1 in June 1979.

Fig. 3.
Dry density and moisture profiles at CP6-1 in June 1979.

Fig. 4.
Dry density and moisture profiles at CPS-1 in June 1979.

B. Geologic Studies at LASL Waste Disposal Sites [M. A. Rogers and B. W. Burton]

All waste disposal pits and shafts at Los Alamos are excavated in the Bandelier tuff, a series of volcanic ashflows and ashfalls on the Pajarito Plateau with a total maximum thickness of 305 m. Geologic descriptions of the structure and stratigraphy of the tuff are essential to a thorough understanding of possible migration pathways from buried waste.

Subunits of the Tshirege Member of the Bandelier tuff can now be correlated across the Pajarito and Los Alamos faults. Estimates of displacement on these faults have been based on topographic relief. More precise values are now possible. Additional information on erosion rates for the tuff may also be gained from this study.

New map subunits for the Tshirege Member of the Bandelier tuff were established for the Pajarito Plateau in the LASL environs. These subunits cover all of the Tshirege cropping out in the LASL environs and enable correlation of the tuff exposed at the distal end of the plateau with tuff exposed at the proximal end. All of the new subunits have recognizable lithologies and/or depositional boundaries, which are traceable across the area of the plateau in which any subunit crops out. The depositional boundaries between subunits are not the only ones present. For example, Unit E has at least four separate ashflows. The four flows are similar in lithology and differ in lithology from the flows in Units F and D above and below.

Old Units 1b and 2a are combined in new Unit B because Unit 1b cannot be distinguished from Unit 2a from midplateau to the mountain front.

To aid in correlation of the units, a test of one sample each from Units D, E, and F were analyzed by high-magnification reflected-light microscopy and by scanning electron microscopy (SEM). The reflected-light microscope revealed textural differences among the three units at

<u>New Units</u>	<u>Previous Working Units</u>
Unit A	Unit 1a
Unit B lower	Unit 1b
upper	Unit 2a
Unit C	Unit 2b
Unit D	Unit 3
Unit E	Unit 4
Unit F	Unit 5

high magnification. The SEM identified some mineralogical differences in the sample from Unit F. It seems to contain a higher concentration of pyroxenes and other unidentified mineral grains than the samples from Units D and E. Some of these unidentified minerals contain considerable cerium and titanium. Whether these textural and mineralogical differences can be said to be typical of each unit would require a much larger study.

From these geologic studies, a new geologic map of the LASL reservation will be made that will furnish far more detail than any previous one. The map will be useful in present and future waste management decisions.

C. Trace Element Study of the Bandelier Tuff

[M. A. Rogers and B. W. Burton]

The rhyolitic Bandelier tuff is divided into a lower and upper member—the Otowi and Tshirege, respectively. Locally, the Otowi and Tshirege are separated by Cerro Toledo Rhyolite tuffs. This study was initiated to find a means to correlate the ash flows in the Tshirege, which is the unit used for waste disposal at LASL.

One hundred twenty samples were collected from 56 locations on the Pajarito Plateau. The majority of the samples came from the Tshirege. The few samples collected from the Otowi and Cerro Toledo Rhyolite tuffs were for general comparison with the Tshirege. The samples were analyzed for zinc, uranium, tungsten, thorium, cesium, rubidium, cobalt, tantalum, iron, barium, antimony, arsenic, and molybdenum. Zinc, uranium, thorium, cesium, rubidium, and tantalum show trends that decrease from bottom to top for both the Otowi and Tshirege. In the Tshirege, tungsten and antimony decrease in the lower part and increase in the upper. The Otowi shows a decrease in tungsten and antimony from bottom to top. Cobalt and iron show trends that increase from bottom to top for both the Otowi and Tshirege. The distribution of barium, arsenic, and molybdenum cannot be easily characterized. With the exception of tungsten and antimony, multiple core samples from one location showed the same results as samples from scattered locations on the plateau. The core samples showed higher values for tungsten and antimony than the outcrop samples.

Statistical analyses, cluster analysis, and discriminant analysis showed that for correlation purposes, uranium, cesium, rubidium, and tantalum, followed by molybdenum and iron, were the most important elements in characterizing the Tshirege.

PRESENTATION

B. W. Burton and M. A. Rogers, "Distribution of Some Trace Elements in Bandelier Tuff," Los Alamos Scientific Laboratory seminar, May 11, 1979.

D. Hydraulic Conductivity Studies at LASL Waste Disposal Sites

[W. V. Abeele]

Surface water and precipitation can infiltrate waste pit overburden, and if present in sufficient quantities, can reach waste material. Water movement from the pit might transport radionuclides from it. Monitoring of the soil moisture at LASL disposal sites indicates that the water content by volume may not reach 6% at depths exceeding 5 m. The rate at which water and dissolved or suspended radionuclides *can* move in unsaturated conditions decreases rapidly as a function of moisture content.

Measurement of unsaturated hydraulic conductivity is difficult. Because of this difficulty, several methods have been suggested for the computation of unsaturated conductivities based on pore-size distribution. The pore-size distribution data can be deduced during simultaneous measurement of matric potential and soil water content.

A proposed method for measuring unsaturated hydraulic conductivity specific to soils was improved by Millington and Quirk (MQ).¹ This derived conductivity function is obtained using numerical methods. Campbell² developed a functional relationship between water content and conductivity that can be used directly in equations describing unsaturated water flow in soil. The differences in conductivity determined by the more intricate³ direct measurement technique [$k(\theta)$ in Table I], and the predictive methods based on the matric potential-water content relationship rarely exceed one order of magnitude in crushed Bandelier tuff. This has been verified in Table I. The relationship between water content and matric potential is also presented

TABLE I
HYDRAULIC CHARACTERISTICS OF BANDELIER TUFF
AS A FUNCTION OF θ_v
(Hydraulic conductivities are in m/yr)

θ_v	ψ	$k(\theta)$	MQ	Campbell
0.40	0	28	28	28
0.16	33	0.11	0.05	0.02
0.14	46	0.07	0.02	7×10^{-3}
0.12	76	1.6×10^{-2}	5×10^{-3}	2×10^{-3}
0.11	100	7.9×10^{-3}	2.6×10^{-3}	9.5×10^{-4}
0.10	133	3.8×10^{-3}	1.3×10^{-3}	4.1×10^{-4}
0.09	184	1.6×10^{-3}	5.4×10^{-4}	1.6×10^{-4}
0.08	264	6.9×10^{-4}	2.2×10^{-4}	6.7×10^{-5}
0.07	397	2.4×10^{-4}	8.0×10^{-5}	2.3×10^{-5}
0.06	636	7.3×10^{-5}	2.3×10^{-5}	7×10^{-6}

in Table I, where θ_v represents the water ratio by volume and ψ the matric potential expressed in kilopascals. A θ_v of 0.40 represents saturation. The predictive methods, mainly the MQ method, can consequently be used to determine the unsaturated hydraulic conductivity.

REFERENCES

1. R. J. Millington and J. P. Quirk, "Permeability of Porous Solids," *Trans. Faraday Soc.* **57**, 1200-1207 (1961).
2. G. S. Campbell, "A Simple Method for Determining Unsaturated Conductivity from Moisture Retention Data," *Soil Sci.* **117**, 311-314 (1973).
3. W. B. Abeele, "Determination of Hydraulic Conductivity in Crushed Bandelier Tuff," Los Alamos Scientific Laboratory report LA-8147-MS (November 1979).

E. Moisture Monitoring

[M. Wheeler]

1. Introduction. During the past several years, monthly measurements have been taken of the water content in the tuff or fill above, beside, and beneath waste disposal pits and shafts at LASL. Concurrently, measurements were made of the hydrologic properties of the tuff (Sec. III.D). The purpose of these measurements is to quantify the magnitude of moisture movement within and below buried waste. Such water movement is considered to be an important potential mechanism for the migration of radionuclides in buried waste.

The moisture content of the earth surrounding an access hole is determined with a neutron moisture gauge. All of the data collected at LASL indicate partially saturated conditions, with maximum values of about 20% by volume (50% of saturation). Data for three representative holes at Area G, TA-54, are discussed below, covering the period May 1978 through November 1979.

2. Access Hole GP1-1. This hole is in crushed tuff overlying waste material. The moisture data (Fig. 1) in conjunction with the hydrologic properties of crushed tuff (Sec. III.D) were used to calculate the moisture flux in the 2- to 3.5-m depth zone. The calculated downward flux varies from 0 to 8 mm/month, with an average of 2.5 mm/month. Should this rate continue for many years, the moisture content of the crushed tuff in the waste pit would rise to about 15% — just sufficient to conduct this flux. (Current moisture content at the top of the waste is about 12%.)

3. Access Hole GS50-5. This is an open hole in solid tuff located about 2 m from a vertical disposal shaft. As shown in Fig. 2, the highest water content was consistently at a depth of about 2 m and never exceeded 20% by volume. Below 2 m, the moisture profile is influenced by downward moisture flow and by the varying hydrologic properties of the tuff. The low values at about 4 m are associated with a pumice bed, which tends to hold less moisture than adjacent material. Below 10 m, the water is nearly constant at a value of about 4%. Tuff at this water content conducts water at a rate of less than 10^{-6} m/yr, indicating there is little, if any, downward moisture flow.

4. Access Hole GP7-3. This hole is cased through backfill and waste material; below the waste it is an open hole in solid tuff. Measurements were not made in the waste because of the influence of waste material on the readings. From the bottom of the pit, at 8 m, the moisture content decreases to a depth of 15 m, and then remains nearly constant (Fig. 3). The tuff at this water content (3 to 4% by volume) will conduct water at a rate of less than 10^{-6} m/yr.

Fig. 1.

Moisture Hole GP1-1 (Month 5 through 12 — May through December 1978; Months 13 through 23—January through November 1979).

Fig. 3.

Moisture Hole GP7-3 (Months 5 through 12— May through December 1978; Months 13 through 23—January through November 1979).

Fig. 2.

Moisture Hole GS50-5 (Months 5 through 12—May through December 1978; Months 13 through 23—January through November 1979).

5. Conclusions. These moisture data show consistently small amounts of moisture penetration to depths below a few meters. Fluxes are larger in crushed tuff overlying waste than in solid tuff beneath waste, suggesting a gradual increase in moisture contents. Further measurements are being made to determine the dynamics of this process and to evaluate the significance of these fluxes with respect to potential migration of radionuclides.

F. Tritium Migration

[M. Wheeler and T. Gunderson]

Vertical shafts augered in Bandelier tuff are used at LASL for disposal of tritiated wastes. Tritiated water vapor from these shafts has been observed to migrate into the surrounding tuff from both unlined shafts and asphalt-coated shafts.^{1,2} In 1976, special tritium waste packaging requirements were implemented. These requirements specified various degrees of encapsulation in asphalt, using 210-ℓ steel drums as the final container. These wastes were placed in a new disposal shaft (Shaft 150), located in an area where no previous tritium disposals had occurred, thus permitting an evaluation of the effectiveness of the new procedures.

The first disposals to Shaft 150 occurred on May 12, 1976, and the shaft was filled (capped with concrete) in the fall of 1979. In the winter of 1977, after more than 31 500 Ci of tritium had been disposed in the shaft, nine sampling holes were augered around the shaft. Tuff samples were processed to remove the water, which was analyzed for tritium. In the fall of 1979, samples of the water vapor from the air in the access holes were collected using silica gel as an absorbing medium. Water was distilled from the gel and analyzed for tritium by liquid scintillation counting. Moisture content in the tuff surrounding the access holes was determined using a neutron moisture probe.

In 1979, average water content in the tuff surrounding Shaft 150 was about 4% by volume. These values are similar to those measured in other parts of the disposal area.

The measured tritium concentrations were converted to volumetric concentrations in the tuff surrounding the shaft, as presented in Table I. From these data, total tritium in tuff surrounding the shaft to a radius of 15 m was computed as shown in Table II.

TABLE I
TRITIUM CONCENTRATIONS IN TUFF
ADJACENT TO SHAFT 150, AREA G

Access Hole	Distance from Shaft (m)	Mean Volumetric Tritium Concentration (pCi/m ³)	
		1977	1979
3	2.1	5.2 ¹⁰	1.0 ¹¹
2	4.0	2.2 ¹⁰	1.2 ¹¹
1	10.7	3.3 ⁹	2.2 ¹⁰
4	2.1	3.8 ¹⁰	1.3 ¹¹
5	4.0	1.4 ¹⁰	1.2 ¹¹
6	10.7	2.8 ⁹	4.4 ¹⁰
7	2.1	1.3 ⁹	1.0 ¹¹
8	4.0	1.0 ¹⁰	1.2 ¹¹
9	10.7	1.0 ¹⁰	4.4 ¹⁰

TABLE II
FRACTIONAL RELEASE OF TRITIUM FROM SHAFT 150

	Tritium Disposed to Shaft (Ci)	Tritium in Tuff Around Shaft (Ci)	Per Cent Leached
Jan. 1977	31 500	150	0.5
Oct. 1979	130 700	930	0.7

Measurements made during two former investigations^{1,2} showed a release of 0.3 to 0.6% of the tritium in the shaft to the surrounding tuff. The present study showed values of 0.5 to 0.7%. It is apparent that no substantial improvement in tritium containment was obtained by the new packaging requirements. It is difficult to establish whether the packaging procedures themselves were inadequate, or whether the problem derived from inadequate quality control.

There is very little, if any, movement of water from the surface of the disposal area to the regional aquifer. Estimated travel times are in thousands of years,³ allowing for complete decay of the tritium. Further measurements are in progress to determine the significance of tritium movement into the air around the shaft.

Results of this study led to use of asphalt-coated steel liners inside an augered shaft. The shafts will be equipped with monitoring pipes external to the liner.

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G. Thorium/Uranium Ratio Study of the Bandelier Tuff [B. W. Burton]

To compare outcrop data from the trace element study (Sec.III.C) of the Bandelier tuff with subsurface data, two samples from each of five horizontal cores taken beneath Pit 3, Area G, TA-54¹ were analyzed for zinc, uranium, tungsten, thorium, cesium, rubidium, cobalt, tantalum, iron, barium, antimony, and molybdenum. At each hole one sample was taken directly from under the pit and one from about half-way between the pit and the hole collar (Table I). A one-way analysis of variance shows no difference at the 99% confidence level between the core and outcrop samples for any element except tungsten and antimony (Table II). The higher concentrations of

TABLE I
THORIUM/URANIUM RATIOS AND LOCATIONS OF
SUBSURFACE SAMPLES IN UNIT C, TSHIREGE

<u>Hole</u>	<u>Sample</u>	<u>Th/U</u>	<u>Downhole Depth (m)</u>	<u>Horizontal Distance from Rim (m)</u>	<u>Depth Below Pit Floor (m)</u>	<u>Depth Below Surface (m)</u>
P-3 MH-2	Under pit	4.2	70.1	68.0	2.7	12.8
P-3 MH-2	Away from pit	3.9	35.0	30.5	Pit level	8.2
P-3 MH-3	Under	4.1	69.2	69.2	2.0	11.9
P-3 MH-3	Away	3.8	40.2	40.2	Pit level	8.8
P-3 MH-4	Under	4.2	66.3	62.2	3.1	12.5
P-3 MH-4	Away	3.9	39.0	36.6	Pit level	8.5
P-3 MH-5	Under	3.9	69.2	53.3	0.01	9.4
P-3 MH-5	Away	4.0	36.3	27.7	Pit level	7.6

these latter two elements in the cores may be partly the result of erosion of the drill bit and partly the result of improvements in analytical techniques between the time the outcrop samples and core samples were analyzed.

A comparison of the thorium/uranium (Th/U) ratios between the core and outcrop samples suggests there may be a redistribution of these elements within the tuff. Table I shows the Th/U ratios and locations of the core samples in Unit C of the Tshirege.* Data from Hole P-3 MH-1 were not used because it crossed one and probably two unit boundaries. The mean of the Th/U values for the eight core samples is 4.04 ($\sigma = 0.17$), and the mean for the corresponding 13 outcrop samples is 3.52 ($\sigma = 0.27$). The difference is significant at the 99% confidence level.

The Th/U ratios in the cores seem to be a function of both horizontal distance from the mesa rim and depth below the original surface of the mesa (Table I). It seems probable that there has been some redistribution of natural uranium relative to thorium in the Tshirege by water movement through the tuff on a geologic time scale. More data are needed for verification and will be collected in the coming year.

*The upper member of the Bandelier tuff.

TABLE II
TRACE ELEMENT CONCENTRATIONS FOR UNIT C OF THE TSHIREGE FROM
OUTCROP AND SUBSURFACE SAMPLES

Element	Outcrops			Cores		
	Number of Samples	(ppm)		Number of Samples	(ppm)	
		Mean	Standard Deviation		Mean	Standard Deviation
Zn	13	72.4	10.4	9	74.4	12.3
U	13	5.0	0.6	9	4.8	0.3
Th	13	17.4	1.7	9	19.2	1.1
W ^a	13	2.8	0.8	9	3.9	0.7
Cs	13	2.9	0.5	9	3.0	0.5
Rb	13	159	19	9	156	12
Co	13	0.79	0.25	9	0.62	0.22
Ta	13	4.2	0.5	9	4.4	0.4
Fe	13	1 x 10 ⁴	1 x 10 ³	9	1.1 x 10 ⁴	1 x 10 ³
Ba	13	302	112	9	252	56
Sb ^a	13	0.24	0.08	9	0.36	0.13
As	13	1.5	0.4	9	1.4	0.3
Mo ^b				9	1.9	0.7

^aSignificantly different.

^bSamples being analyzed.

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H. Radioactivity and Hydrologic Waste Management Field Studies in Tuff at Los Alamos [J. W. Nyhan and G. Trujillo]

The LASL Environmental Science Group is conducting field-oriented nuclear waste management research related to radionuclide and water migration in volcanic tuffs. As part of this effort, a field-drilling operation was conducted in Los Alamos during 1978 to sample tuff beneath a previously used liquid radioactive waste disposal site (Area T). Continuous split-spoon tuff samples were collected every 15 cm to a depth of about 31 m in two 36.6- by 6.1-m absorption beds (pits) at this site.¹ Tuff samples collected were assayed for their water content, and these hydrologic data were then compared with preliminary geologic data collected in these two holes (Fig. 1). A few preliminary transuranic analyses were also performed on these samples (Table I).

The relatively high water contents of the tuff samples described in Fig. 1 reflect the fact that absorption bed 1 received large quantities of liquids in 1961 (about 800 000 l of tap water and about 675 000 l of transuranic effluents that absorption bed 2 did not receive) in an experimental effort to change the original vertical distribution of transuranics beneath absorption bed 1. Our

TABLE I
NET TRANSURANIC RADIONUCLIDE COUNT
RATES IN AREA T TUFF SAMPLES

Depth (m)	Radionuclide Net Counts/s*	
	Plutonium	²⁴¹ Am
Absorption Bed 2, Hole 1		
2.44-2.59	28.59	118
3.20-3.35	1.67	10.8
3.66-3.81	1.93	10.8
6.10-6.25	0.456	3.49
9.14-9.30	0.355	2.46
12.19-12.34	0.0129	0.0044
15.24-15.39	0.005	0.0
18.29-18.44	0.0169	0.0
24.69-24.84	0.0155	0.0039
30.33-30.48	0.0229	0.0
Absorption Bed 1, Hole 1		
2.36-2.44	2.03	13.9
3.20-3.35	80.3	401
3.66-3.81	25.7	138
6.10-6.25	65.9	353
9.14-9.30	14.8	74.6
12.19-12.34	1.89	3.76
15.24-15.39	0.299	1.06
18.29-18.44	0.613	2.30
24.38-24.54	0.374	1.24
30.33-30.48	0.419	2.48

*Numbers represent net counts/s in approximately 25 g of dry tuff.

most recent information on the tuff below absorption bed 1 was compared with similar information collected in 1961 immediately before and after the addition of water and effluents to the absorption bed. This comparison shows that the water added to absorption bed 1 in 1961 has continued to move downward, even to a depth of 31 m, as evidenced by the elevated water content of tuff samples collected in absorption bed 1 vs absorption bed 2 (Fig. 1).

The preliminary radionuclide data for a few selected tuff samples from each of the two absorption beds studied at this site are presented in Table I. These data are presented as net counting rates for these samples because the calibration work on the ATASS system (an improved automatic transuranic assay system for tuff and soil samples) is not complete as yet (Sec.III.I). The first sample described for each absorption bed in Table I represents samples collected immediately above the layer of stones and gravel lining the bottom of the original absorption bed (2.44- to 2.59-m sample in bed 2 and 2.36- to 2.44-m sample in bed 1); the second sample listed describes the radionuclide data for the sample collected immediately beneath the bottom of the bed (3.20- to 3.35-m samples in both beds). It is not surprising that these four tuff samples exhibited high concentrations of transuranics because the first sample is actually part of the old

Fig. 1.

Geologic logging data and water content of tuff samples as a function of depth for two absorption beds at Area T.

absorption bed. Detectable levels of radionuclides appear as far down as the 9.14- to 9.30-m samples collected in absorption bed 2, and as far down as the 30.33- to 30.48-m samples in absorption bed 1. Our preliminary results thus indicate that significant downward migration of americium and plutonium, as well as water, has occurred since 1961 in absorption bed 1 at Area T, an observation that has significant long-term implications for nuclear waste management and hydrologic research.

Work to be performed in FY 1980 will continue to elucidate more precisely to what depth this migration of radionuclides and water has occurred and exactly what total amounts of radionuclides are present. Correlations will also be made between the vertical migration of radionuclides and stable element components of the waste solutions into the tuff.

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I. An Automated Radioactivity Assay System for Soils

[J. W. Nyhan, G. Trujillo, B. Drennon, and J. Crowell]

Assaying soil and tuff samples containing low concentrations of transuranics currently requires time-consuming, costly, and highly specialized analytical procedures. Currently, soil samples are dissolved in concentrated acid solutions and then passed through an ion exchange resin to achieve chemical separation. The sample is then electroplated onto a metal disk and assayed for transuranics using alpha spectrometry techniques. These procedures take several weeks and cost about \$200/sample. An automatic transuranic assay system for soils (ATASS) that reduces the time and expense of analyzing transuranics in soil and tuff samples from radioactive shallow land burial sites has been developed (Fig. 1). The ATASS simultaneously measures the low- and high-energy gamma spectra (0 to 2000 keV) of the components of soils and crushed geologic materials. An evaluation of the spectrum leads to the quantitative identification of the transuranic and nontransuranic sample constituents.

The ATASS counting system includes two germanium detectors that simultaneously assay a sample. The intrinsic germanium (IG) detector (Fig. 2) consists of a hyperpure germanium crystal with a thin metal semiconductor surface barrier entry window. The IG crystal is mounted in a cryostat, which has a thin beryllium window and a cooled field effect transistor (FET). This detector has excellent photon-peak resolution in the 0- to 200-keV range with a total active detector volume of 14.7 cm³ and a crystal thickness of 7 mm. In order to also assay for high-energy (200 to 2000 keV) gamma emitters with ATASS, a coaxial Ge(Li) detector was added to the system. The second detector (Fig. 2) has a right angle detector-dewar configuration and a total active volume of 125 cm³. These two detectors are interfaced with analog-to-digital converter (ADC) multiplexers, pulse amplifiers, high-voltage power supplies, a Canaberra model 8100 multichannel analyzer (MCA), and a PDP-11/04 minicomputer.

The ATASS is mounted in a sample changer constructed to accept especially designed plastic sample containers. The sample changer consists of a lead-lined wheel, approximately 0.91-m diam and 2.5 cm thick, which is mounted vertically in the center of an aluminum framework (Fig. 3). The wheel has 20 evenly spaced sample holder positions. A small motor mounted in the center of the wheel allows it to turn and position a sample between the two detectors. The wheel holds the plastic sample containers, which are 7.1-cm-diam by 0.74-cm-thick and were designed to hold approximately 25 g of soil or tuff. The lid of the sample container is less than 1 mm thick and faces the IG detector (where low-energy gamma emitters are assayed), whereas the bottom of the sample container is twice as thick as the lid and faces the Ge(Li) detector for high-energy gamma emitter assays.

Although additional system characterization work is still needed, preliminary indications are that ATASS is a very effective, inexpensive radionuclide assay system for waste management research. The sensitivity of the IG detector is demonstrated by uncontaminated tuff samples spiked with weapons grade plutonium and americium standard solutions to mimic field samples with activity levels of 50 pCi/g (Fig. 4). Notice the good peak resolution in the low-energy L x-ray region (0 to 20 keV) and that the spiked sample spectra are clearly represented above the natural background sample, which defines peaks of naturally occurring elements such as ²¹⁰Pb, ⁴⁰K, and thorium.

The ATASS system was calibrated for ²⁴¹Am detection and plutonium calibration work was initiated. Coefficients for converting counts per second to picocuries ²⁴¹Am/g have been determined for the 59.537-keV gamma ray and for the americium x rays (Table I). Furthermore, the ratios of x-ray intensities to gamma-ray intensity were determined so that the americium contribution of the x rays can be stripped from the spectra for plutonium analysis. We have also examined linearity of response of the IG detector to varying low radionuclide concentrations have found that the IG detector does respond linearly with increased concentrations of americium placed in

empty sample containers (Table II). The results of similar experiments with weapons-grade plutonium are shown in Table III; however, the large amount of variation between replicate plutonium assays in this experiment has led us to suspect the homogeneity of the plutonium standard solutions and preclude any conclusions about linearity of response for plutonium at this time. We expect to measure activities as low as 5 pCi Pu/g and 0.05 pCi ^{241}Am /g with maximum sample counting times of 4.5 h.

Fig. 1.

An automated transuranic assay system for soils (ATASS).

Fig. 2.
ATASS detectors. (a) IG detector and cross section of detector element; (b) Ge(Li) detector and cross section of detector element.

Fig. 3.
Illustration and two views of ATASS automated sample changer.

Fig. 4.
IG Spectra.

TABLE I
RADIONUCLIDE SENSITIVITY CALIBRATION
FOR ATASS SYSTEM

			$0.0161 \frac{\text{pCi Am}}{\text{pCi Pu}}$
²⁴¹ Am Content in ²³⁹ Pu Standards			Sensitivity (counts/s)/(pCi/g)
²⁴¹ Am	59.537 keV	γ ray	0.0283 ± 1.6%
	L _α	x ray	0.00137 ± 2.5%
	L _β	x ray	0.00542 ± 1.3%
	L _γ	x ray	0.00179 ± 1.2%
²³⁹ Pu ^a	L _α	x ray	0.000315 ± 4.6%
	L _β	x ray	0.000970 ± 3.7%
	L _γ	x ray	0.000273 ± 2.8%

^aCorrected for ²⁴¹Am content.

TABLE II
LINEARITY OF IG DETECTOR RESPONSE TO
VARYING AMOUNTS OF ²⁴¹Am

²⁴¹ Am Added to Sample Container (pCi)	Average Net Counts for 16 000-s Count Time
1.02	37.0 (33.5) ^a
5.10	139 (7.02)
10.2	258 (21.5)
102	2 513 (220)
1 020	28 360 (4606)
15 300	427 237 (5384)

^aNumbers represent average of three replicate determinations and standard deviation of mean.

TABLE III
IG DETECTOR RESPONSE TO VARYING
AMOUNTS OF WEAPONS GRADE
PLUTONIUM

Plutonium Added to Sample Container (pCi)	Average Net Counts for 16000-s Count Time
4.5	102 (68.8) ^a
9.0	168 (95)
19.5	91 (66.6)
22.5	112 (71)
45	265 (69.9)
90	392 (93)
1 500	4 103 (239)
13 320	45 621 (2972)

^aNumbers represent average of three replicate determinations and standard deviation of mean.

J. An Investigation of Alternatives to Shallow Land Burial for the Disposal of Nontransuranic Low-Level Wastes*

[Compiled by M. Wheeler]

Increasing concern over the adequacy of shallow land burial for disposal of low-level radioactive waste has focused attention on the need to develop alternatives.¹ This program, initiated in CY 1978, is designed to describe the technical requirements of selected alternatives and to identify current technology gaps. The alternatives include various geologic options, engineered storage, and ocean disposal.

1. Mines. An in-depth investigation of operating and abandoned mines² revealed that (1) most abandoned or inactive mines are generally in very poor condition, both structurally and hydrologically, and offer little possibility for waste disposal, (2) dual purpose mines (mining and waste disposal) would be possible, but would have to be planned from the beginning of excavations, (3) nonoperational salt mines and underground limestone quarries offer the best possibilities for conversion to waste disposal facilities. Preliminary estimates were made of the cost of mine conversion. Although higher than for current land burial, the increased costs may be offset by increased security and reduced long term care.

2. Mined Cavities. Cavities excavated specifically for waste disposal can have many advantages over converted facilities, including tunnel stability, waste handling considerations, reduced impact on the hydrology of surrounding rocks, and siting and access considerations.³ This program details existing knowledge of rock mechanics and hydrology, which can be applied to the development of mined-cavity disposal.⁴

3. Natural Cavities. A survey was made of the types of natural cavities, and their geologic, hydrologic, and physical aspects were described.² Without exception, these cavities were determined to be unsuitable for waste disposal.

4. Engineered Storage. Above- or below-grade structures engineered for waste storage offer opportunities to rigidly control the disposal environment. Knowledge concerning the long-term behavior of such structures, and likely failure modes, has been obtained from studies of ancient caves and burial vaults.⁴ Egyptian and Chinese tombs dating back several thousand years have provided excellent preservation of stored material, except as they were breached by treasure hunters. Similar structures might be used to isolate waste materials.

5. Ocean Disposal. Techniques for ocean disposal were catalogued, including direct dumping, sea bed placement, and undersea tunneling.⁴ All such operations will be greatly complicated by existing treaties and international agreements, as well as by technical and environmental problems. Further studies are in progress.

6. Applicable Research. Specific technical uncertainties are being identified for each alternative. Many of these, particularly those pertaining to geologic alternatives, are being addressed by research in other areas. Contact has been established with the Office of Nuclear Waste Isolation (ONWI) to identify such programs under its jurisdiction. Opportunities for mutually beneficial research will also be identified.

*A summary of the research directed by Professor M. E. Wacks, University of Arizona for the LASL Group LS-6 program, "Radioactive Waste Burial Technology."

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K. A Study of the Shallow Land Burial of Low-Level Radioactive Wastes

[Compiled by J. W. Nyhan]

Work done by the University of Texas at Austin during FY79 involved engineering and hydrologic research in support of LASL studies of radioactive waste burial technology. Major activities have involved (1) identification and evaluation of potential liner materials for disposal pits, (2) development and proof-testing of experimental apparatus for measuring hydraulic conductivity of compacted soil barriers, (3) refinement of a program for analyzing water movement and containment times for soil barriers, and (4) assessment of problems associated with settlement of wastes in disposal pits.

Based on a preliminary study of liner materials used in various shallow land burial configurations, liner service lives (as moisture barriers) and costs have been estimated (Fig. 1). Life expectancies for liners to be used to retain radioactive waste solutions were also estimated for some liner materials. Past experience indicates that synthetic liners, in general, have an expected life of around 25 yr. Although some research is underway in several DOE laboratories to determine the long-term effects of different chemicals and leachates on synthetic liners, these experiments have been in progress for less than 3 yr. This is far too short a time to provide data on long-term stability needed for disposal of low-level radioactive wastes. Although synthetic liners may provide a short-term solution to containment of radioactive wastes, they do not at present provide a cost-effective containment even for the short run.

Concrete and asphalt are more viable prospective liners than are synthetic membranes. At present, asphalt appears to contain radioactive waste solutions more effectively for a given cost than concrete. However, man made materials are susceptible to degradation by environmental factors or active chemicals in the waste (Table I).

We conclude that thick, compacted clay liners offer the most inexpensive barrier for a given containment time of the waste. Thicknesses of 2 to 3 m (7 to 10 ft) will retard outmigration of elements in solution. Cracking can be minimized by proper selection of compaction density and water content, possibly with the help of admixtures. Spray-on asphalt membranes will reduce and possibly eliminate cracking during construction.

Fig. 1.

Comparison of expected service life vs in-place cost (1976). Unless noted, service life estimate is for water-retaining structures. A + is the radwaste estimate; ++ means no estimate is available (because the liner was invented in 1975). Cost estimate includes excavation, installation, backfilling, compaction, and seeding (glacial till). A # means the cost does not include construction of subgrade or earth cover.

To measure the hydraulic conductivity of compacted soil barriers of the type that might be used at Los Alamos or some other site in an arid region, an experimental method that employs thermocouple psychrometers in conjunction with the instantaneous profile method has been developed. For clays, the method works well for degrees of saturation between about 20 and 90%. This range is more restricted with sands, but the problem can be remedied by replacing psychrometers with tensiometers. Measurements of hydraulic conductivity obtained with the psychrometers were used as input to a computer program designed to predict rates of water movement in compacted soil barriers (Fig. 2). Predicted rates were compared with rates measured in laboratory experiments. The comparisons were good and it is therefore concluded that the method of measuring conductivity and predicting moisture movement work well for compacted soil barriers. The purpose of the air gap (Fig. 2d) in the treatment involving Los Alamos tuff plus 15% bentonite was to simulate what happens to the flow of water through tuff when a fracture occurs in the water path. The gap acts as a barrier, as hydrologic theory predicts, to unsaturated water flow in fissured geologic units like Bandelier tuff.

Another area of investigation during 1979 was the reviewing of information available on settlement of municipal landfills and exploring the implications of this information to shallow land burial of low-level radioactive wastes.

TABLE 1

INFLUENCE OF ENVIRONMENT — POLYMERIC MEMBRANES

	<u>Butyl</u>	<u>EPDM</u>	<u>Neoprene</u>	<u>CSPE</u> <u>(Hypalon[®])</u>	<u>CPE</u>	<u>PE</u>	<u>HDPE</u>	<u>PVC</u>	<u>Elasticized</u> <u>Polyolefin</u> <u>(3110)</u>
Resistance to Ozone	Cracks	Good	Good	Good	Good	Poor (should not be exposed)		Poor (should not be exposed)	Good
Resistance to Ultraviolet Rays			Good	uv rays may cause premature curing	Good	Poor (should not be exposed)		Poor (should not be exposed)	Good
Resistance to Deterioration of Properties at Extreme Temperatures	Good	Good	Good	Good	Good; excellent crack re- sistance at low temp	Poor		Poor	Good
Resistance to Soil Microorganisms				Good	Good		Good	Poor	
Resistance to Damage from Rodents	Poor	Poor	Poor			Poor	Good	Poor

(a)

(b)

(c)

(d)

Fig. 2.

Comparison between predicted and measured final water content profiles for (a) compacted Los Alamos tuff, (b) kaolinite, (c) a mixture of Los Alamos tuff plus 10% bentonite, and (d) a mixture of Los Alamos tuff plus 15% bentonite.

Settlement of waste can lead to cracking of earthen covers over pits and to infiltration of water. Settlement is caused by compression of the waste under overburden pressure, flow of parts of the waste into openings at greater depth (raveling), and decomposition of refuse. Settlements may be as much as about 50% of the original thickness of waste, as shown by compression indices ranging from 15 to 55% of the in-place void ratio, depending on the organic content of the wastes (Fig. 3). Much of the settlement occurs during the first few months, but significant movements may continue for years. Settlement may be minimized by packaging the waste in advance, by filling voids with soil, by keeping moisture out of the waste, and by preloading. With proper engineering, expensive solutions to the problem of settlement, such as grouting and continued maintenance of the cap, would be unnecessary.

The work completed during 1979 resulted in (1) development and documentation of several design concepts for an improved disposal pit, (2) some conclusions concerning the best materials to be used as barriers in an improved pit, and (3) advancements in the technology of predicting containment times for compacted soil barriers. With these efforts largely completed, sufficient knowledge and technology now exist to begin designing and evaluating alternative disposal pits for improved shallow land burial of low-level radioactive wastes. The bulk of the work in the upcoming fiscal year will be directed toward developing such engineering designs and improving our understanding of hydrologic relationships in barriers engineered to contain low-level radioactive wastes.

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Fig. 3.

Compressibility of waste as a function of initial void ratio and amount of organic matter.

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L. Solute Retention in Waste Disposal Sites at the Los Alamos Scientific Laboratory [Compiled by J. W. Nyhan]

Movement of solutes through soils and other porous materials is strongly dependent on the ability of the materials to react with and retain the solute constituents. This retention capacity is a critical factor in evaluating the suitability of porous materials as disposal media for various wastes, and is especially important in the shallow land burial of low-level radioactive wastes. When retention data are integrated with hydrologic data for a site, predictions of radioactive solute transport within and beneath the site can be made.

Whereas the reactions of numerous radioactive and nonradioactive aqueous solutes with a variety of porous materials have been studied, such studies tend to be either solute-specific on soils of widely different properties or to be centered on the reactions of many solutes with specific porous materials. With regard to waste management research such as that started last year at

New Mexico State University, there is a critical need for systematic studies of the reactions of radioactive solutes with chemical properties similar to several other chemicals of national interest in soils and geologic materials. If reactions of these chemical analogs are studied in a variety of materials with pertinent properties that vary within reasonable ranges for soils of a particular region, experimental results are much more likely to be applicable at a wider level than most previous studies.

Research performed during FY 1979 at New Mexico State University involved the chemical and physical characterization of the 10 soils selected for study and the determination of the adsorptive behavior of nickel, zinc, and boron on some of these soils. The initial results show a good range in those soil properties expected to affect solute behavior in the two Los Alamos soils, a Los Alamos tuff sample, three soils from waste disposal sites in Albuquerque, and four agricultural soils from around New Mexico.

In the chemical adsorption studies, a basic batch equilibration technique is used, where, following equilibration of soils with pollutants, soil suspensions are centrifuged to separate the soil from the solution and an aliquot taken for solute analysis. The amount of solute adsorbed is then expressed as a function of equilibrium solution concentration by using the Freundlich relationship

$$X/M = KC^N,$$

where X/M is the amount of solute adsorbed per mass of soil ($\mu\text{g/g}$), C is the equilibrium solution concentration ($\mu\text{g/mL}$), and K and N are actually empirical constants but have been related to the bonding energy of some solutes. The Freundlich equation is usually converted to the linear form for easy verification of data conforming to the Freundlich relationship

$$\log (X/M) = (1/N) \cdot \log C + \log K.$$

Adsorption data plotted on log-log paper that yield a straight line conform to the Freundlich relationship, and the "constants" K and N are easily determined from the intercept and slopes of such straight-line plots. An example of a Freundlich plot of nickel adsorption from aqueous solution by Carjo soil and regression-determined adsorption parameters is given in Fig. 1. Freundlich nickel adsorption parameters determined for most of the soils studied are presented in Table I.

In general, nickel was very strongly adsorbed and apparently on high energy, specific adsorption sites. Throughout the 0.01- to 1000-ppm concentration range used, >98% of solubilized nickel was retained, and at concentrations below 10 ppm, >99.9% of the nickel was adsorbed. Preliminary results for Zn indicate similar adsorptive behavior (Table II). Neither metal would be expected to move very far in waste disposal sites relative to less reactive metal species. Also, note that zinc behavior in soils would be analogous to soil interactions with other cationic trace elements that interact strongly with soil organics and that nickel and cobalt are chemical analogs.

Boron was initially selected for study because it is representative of neutral or anionic species that are often mobile in many porous media. Although boron adsorption on the six soils studied to date is rapid, with equilibrium apparently attained within 4 h, boron retention in the same soils was much less than for the metal cations (Table III). Adsorption appeared to be highly correlated with soil texture (clay content). Thus, in sandy soils, boron would be expected to move readily, whereas its movement in clay soils would be greatly retarded.

Fig. 1.

Nickel sorption by Carjo soil series: linearized Freundlich plot of $\log(X/M) = (1/N) \cdot \log(C) + \log(K)$.

TABLE I

FREUNDLICH ADSORPTION PARAMETERS AND CORRELATION COEFFICIENTS FOR Ni^{2+} ADSORPTION ON EIGHT SOILS

Material	Using All Data Points			Omitting Points for $C_{\text{initial}} = 1000 \text{ ppm}$			Omitting Points for $C_{\text{initial}} = 100 \text{ and } 1000 \text{ ppm}$		
	K	1/N	r^2	K	1/N	r^2	K	1/N	r^2
Carjo	39.5	0.74	0.979	77.9	0.85	0.991	159	0.95	0.999
Tuff	2.83	0.77	0.982	3.76	0.86	0.994	4.93	0.92	0.997
R-28	47.4	0.75	0.953	187	0.96	0.996	298	1.02	0.999
R-30	28.6	0.71	0.958	75.6	0.87	0.984	214	1.01	1.000
Glendale	117	0.87	0.976	341	1.03	0.999	332	1.03	0.999
Reagen	97.7	0.86	0.978	256	1.01	0.999	279	1.02	0.998
Doak	34.0	0.79	0.977	72.1	0.92	0.996	115	0.98	0.999
Naff	58.2	0.76	0.979	125	0.88	0.993	236	0.96	0.999

TABLE II
AVERAGE PER CENT ADSORPTION OF ZINC FROM A 100-ppm
ZINC SOLUTION AS A FUNCTION OF TIME

Equilibration Time (h)	% Adsorption ^a			
	Glendale	Carjo	Puye	Tuff
1	99.9	98.4	95.5	80.6
3	100.0	99.1	95.2	88.3
6	100.0	99.2	95.4	70.0
12	100.0	99.4	96.1	75.7
24	100.0	99.5	96.4	88.6
48	100.0	99.5	97.7	97.3
72	100.0	99.7	98.2	97.2

^aAverage of three replicates.

TABLE III
FREUNDLICH ADSORPTION PARAMETERS AND CORRELATION
COEFFICIENTS FOR BORON ADSORPTION ON SIX SOILS

Soil	Using All Data Points			Omitting Points for $C_{initial} = 1 \text{ ppm}$			Clay Content (%)
	K	1/N	r ²	K	1/N	r ²	
Glendale	2.75	0.82	0.952	4.03	0.67	0.991	57.0
Reagen	2.24	0.67	0.986	2.79	0.62	0.991	27.3
Carjo	1.74	0.64	0.996	----	----	----	25.0
Puye	0.64	0.73	0.998	----	----	----	10.0
Chem Bottom	0.53	0.69	0.981	----	----	----	5.0
Tuff	0.23	0.76	0.986	----	----	----	3.4

Adsorption studies with arsenic, boron, strontium, zinc, and nickel on all 10 soils will continue during FY 1980. Reactions of nickel, zinc, and boron will be emphasized, with particular attention given to nickel and zinc competition for high-energy sites in soils. Column studies of solute mobility under unsaturated conditions will also be initiated, and tritiated water will be used in these laboratory experiments to determine the water flow characteristics of the experimental soils and tuff.

PUBLICATION

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IV. TRANSURANIC WASTE MANAGEMENT STUDIES

A. Low-Density Waste Assay: MEGAS II

[D. A. Close]

The MEGAS II assay system (see Fig. 1) has been completed and is ready for field trials. The system assays 60-l-volume packages of combustible wastes at detection levels below 10 nCi/g. The system has three separate detection components: a large-area NaI detector, a planar hyperpure germanium detector (both for photon detection), and four ^3He neutron detector banks. The

Fig. 1.

MEGAS II with the high-resolution hyperpure germanium photon detector (rear center), NaI photon detector (inside lead shielding to the right), the cavity inside the ^3He neutron detectors, and the data acquisition and analysis system (on the left and far right).

system had been expected to be installed in an assay room at the CMR Building at LASL, but because of construction problems, this room is not ready for occupancy. Therefore, the MEGAS II system was moved to the old DP Site at LASL and will begin operation trials at the beginning of CY 1980 as part of a D & D program under way at that location.

B. High-Density Waste Assay: 55-Gal Drum Counting Via ^{252}Cf Irradiation

[T. W. Crane]

A combination active/passive nondestructive assay (NDA) system is being developed for the measurement of a wide range of plutonium-contaminated solid wastes. The waste, packaged in 55-gal (208 liter) barrels, could be of either high or low density, with gamma-ray radiation from fission products as high as 1000 R/h. The passive NDA measurement technique consists of counting coincident and total neutrons with ^3He -filled proportional detectors. The active assay, which is made after the passive measurement, employs a cyclical irradiation by a ^{252}Cf neutron source, followed by delayed neutron counting with the source transferred to a shielded storage container. Approximately 1000 s are used for the combination active/passive assay. The three standard deviation detectability limit is near the 10 nCi/g fiducial for the passive plutonium measurement and below the fiducial for the active measurement for barrels with a mass of 100 kg or more. Figure 1 shows the prototype development system whereas Fig. 2 shows the conceptual design for an aboveground shielded system. A belowground system has also been designed. A report on this system¹ has been released. Close liaison has begun with Rockwell Hanford to facilitate a test and evaluation (T & E) program to include the design, construction, and development of such a 55-gal-drum assay system at Richland. The assay system would service all of the various waste generators at that site. Because of the very long time scales for production of such instrumentation (2 yr), it is vital that a T & E program begin as soon as possible.

Fig. 1.

Top view of the experimental arrangement of the active-passive ^{252}Cf shuffler system for the assay of high-density wastes in 55-gal barrels.

Fig. 2.
Proposed combination active-passive ^{252}Cf shuffler assay system for 55-gal barrels.

PRESENTATIONS

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C. High-Density Waste Assay: Large Crate Counting Via Passive Neutron Counting [J. T. Caldwell, H. F. Atwater, and E. R. Shunk]

In the United States a very common transuranic (TRU) waste-storage and transport package is a wooden box having external dimensions of 1.2 m by 1.2 m by 2.1 m. The large physical size of such packages has in the past presented great difficulties for TRU assay and/or screening measurement systems. In addition, it is common practice at many facilities producing TRU waste to band together six of the common 200-l waste barrels into a large "six-pack" package for

convenience in transportation and/or storage. The external dimensions of these waste packages are typically 1.2 m by 1.0 m by 1.8 m, which are again very difficult to assay or screen for TRU activity with currently existing equipment.

We have embarked on a detector-development program aimed specifically at this very large waste package assay/screening problem. The approach we are pursuing is 4π -solid-angle passive neutron counting of the packages. The nominal dimensions of the 4π counting system are 1.4 m by 1.4 m by 2.4 m, sufficient to accommodate either of the common large waste packages mentioned above. Since facilities generate even larger packages, we have pursued a modular approach in the detector design so that a larger (or smaller) detection system may easily be built by using the present design. In fact, in the recent past we have built similar 4π neutron counting systems of dimensions up to 2.4 m by 3.0 m by 6.1 m, intended for use as vehicle portable monitors.¹

The principal isotope-producing neutrons in common TRU waste is ^{240}Pu , although, depending on the chemical form, all alpha-emitting isotopes in the waste contribute to the neutron output of the package. In our system, two types of passive neutron simultaneous measurements are total neutron output and coincidence neutron rate. The latter is a direct measure of the spontaneous fission rate of all fissioning materials in the waste package, whereas the former depends, in addition, on all alpha-emitting material in the package and on the chemical form of the waste.

In typical light water reactor (LWR) fuel waste (with curium isotope removal), the total neutron source for TRU waste in the oxide form is about 300 n/s/g of included TRU. This will increase by about 10% in 5 yr with the grow-in of ^{241}Am . The inclusion of certain low atomic number materials, such as beryllium, boron, lithium, fluorine, etc., will also lead to an increased source term. The spontaneous fission neutron source of this waste is on the order of 200 n/s/g of included TRU, which corresponds to about 100 spontaneous fissions/s/g of included TRU.

The basic module in the system is a rectangular cross-section polyethylene (CH_2) box in which are placed several 1.82 m by 0.05 m by 2 atm fill pressure ^3He proportional counters. We have found that this "neutron chamber" approach makes very effective use of ^3He proportional counters as compared to the usual practice of imbedding proportional counters in a solid CH_2 matrix. One version of our neutron-chamber module consists of a rectangular cross-section chamber constructed of 2.5-cm-thick CH_2 and having external dimensions of 1.22 m by 2.44 m by 0.25 m. Our measured intrinsic neutron detection efficiency for this 2.98 m² module for a bare ^{252}Cf spontaneous fission source is 16% when six 1.82 m by 0.05 m by 2 atm ^3He proportional counters are placed with equidistance spacings within the CH_2 neutron chamber. The same module displays a 15% intrinsic detection efficiency for a ^{252}Cf source moderated by 3 cm of CH_2 .

Optimization studies have revealed that improved performance per ^3He counter is obtained with a two-chambered module. For instance, a module constructed of 1.2-cm-thick CH_2 with a 2.5-cm-thick CH_2 dividing wall displays an intrinsic ^{252}Cf spectrum neutron efficiency of 20% with the six ^3He counters divided between the two chambers. An added feature of this arrangement is that the count-rate ratio between the two chambers is spectrally sensitive and may be used to correct for waste-matrix-generated efficiency modifications.

As pointed out in Ref. 1, a large increase in detection sensitivity in 4π neutron counters is obtained by shielding out cosmic-ray-generated backgrounds. We propose taking advantage of this fact by providing an economic compacted dirt shield for our system. The housing for such an economic system can be made from prefabricated 3-m-diam (or larger) concrete drainage culvert piping, which is typically rated for ≥ 5 -m compacted dirt overburden. Such a 5-m dirt shield, easily implemented at waste burial or storage grounds, results in a factor of 10 improvement in detection sensitivity as compared to the unshielded case.

The current version of our very large package-counting system has a 4π intrinsic detection efficiency of 15 to 20%, depending on the modules used. We use eight of the modules discussed earlier, routing separate inputs to a microprocessor. Our system minimum-detection sensitivity for a typical TRU isotopic mix resulting from LWR-generated waste is ≤ 10 mg of included TRU. This assumes the use of cosmic ray shielding and a 1000-s count time, and is based on obtaining a 5σ above background net signal. The 10-mg sensitivity applies to TRU located anywhere within a 1.2- by 1.2- by 2.1-m waste package. Figure 1 is a conceptual design of an earth-shielded-crate counting system.

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Fig. 1.
End view of detector module of a passive neutron counting, barrel-assaying system.

D. High-Density Waste Assay: Application of Linear Accelerator Technology

[M. R. Cates, L. A. Franks, B. W. Noel, J. L. Pigg, and D. A. Close]

The upper limit of TRU allowed in solid wastes that are put into nonretrievable storage in DOE facilities is 10 nCi/g.¹ In high-density matrices (about 0.3 g/cm³ or greater), the spontaneous photon emission from TRUs is severely attenuated and spontaneous neutron emission (from the ²⁴⁰Pu found with ²³⁹Pu, as an example) is produced at low rates* that can be dominated by natural background counts.^{2,3} Adding the complication of beta-gamma emitters in the matrix further compromises passive counting reliability. We are studying active interrogation techniques, employing an electron linear accelerator (linac), as a part of the LASL waste management instrumentation development program. We have assayed barrels containing only 11 mg (7 nCi/g) of plutonium.

1. Summary of Photon Interrogation Results. Bremsstrahlung photons produced by electron bombardment were used to directly interrogate a 1-ℓ sample can and a 55-gal (208-ℓ) barrel filled with matrix material. Delayed neutrons were counted after 5-min irradiations. Helium-3 gas proportional counters detected neutrons moderated in a polyethylene (CH₂) counting chamber. The delayed neutron yield detected is

$$Y_D = \epsilon R_p \sum_{i=1}^n \beta_i / \lambda_i ,$$

where:

Y_D = total counts,

ϵ = detector efficiency,

R_p = photofission rate (fissions per second),

β_i = neutrons per fission of the i th delayed group

λ_i = decay constant of the i th group, and

n = number of delayed neutron groups.

Figure 1 shows delayed neutron counts from various masses of plutonium in 105 kg of aluminum. We detected less than 10 nCi/g and inferred a detection limit near 1 nCi/g. The electron energy was 18 MeV.

2. Photon and Neutron Interrogation Results in the Same Assay Geometry. A beryllium target was used for some runs to add a fission signature from neutron interrogation. Fission induced by neutrons, in combination with photofission, is useful for separating the assay of fissile materials (like plutonium) from fertile natural constituents like uranium and thorium. Figure 2 shows the ratio of neutron-induced fission to photofission yields from uranium and plutonium samples in an aluminum matrix (105 kg) at different electron energies. The significant increase in the ratio for ²³⁸U above 10 MeV electron energy is because the fast-neutron fission yield increases rapidly with the increased average neutron energy, whereas the photofission yield increase is only gradual. In ²³⁹Pu, the neutron-induced fission is dominated by the low-energy end of the neutron spectrum, and both yields increase at about the same rate.

*For example, in a 100-kg package containing about 15 mg of plutonium (10 nCi/g), the spontaneous neutron emission rate (0.3 to 1/s), multiplied by the detector efficiency (0.1 to 0.25), gives rise to count rates in the range of 0.03 to 0.25/s. Background counting rates for detector volumes ≥ 200 ℓ are typically two or more per second.

Fig. 1.

Delayed-neutron yield from plutonium in 208-l barrel.

Fig. 2.

Delayed-neutron yield ratios from neutron- and photon-induced fission.

3. Other Advantages of Linac Interrogation. Besides the demonstrated sensitivity of linac assay techniques, these systems offer other advantages in waste management applications: (1) compatibility with passive counting systems; (2) energy variability and control of beam current; (3) capability of assaying large packages, such as six packs of waste barrels; (4) ability to scan containers to locate contaminated volumes; (5) accessibility of both neutron and photon interrogation from the same instrument and in rapid sequence, if desired; (6) relative insensitivity of matrix effects for photon interrogation; and (7) applicability of counting and coincidence techniques that are synchronized to the pulse structure of the beam. The continuing LASL program is undertaking applications research in these six areas as well as in advanced detection systems that can exploit the linac active interrogation approach to waste assay.

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PUBLICATION

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E. Fission Gas Detection Development: Resonance Ionization Spectroscopy [M. R. Cates and L. A. Franks]

As part of the accelerator based assay program, LASL devoted some effort this fiscal year to the application of resonance ionization spectroscopy (RIS) for the analysis of transuranic materials contained in solid wastes. The general technique uses laser sources and has been developed at ORNL for detecting very small amounts of certain elements. In collaboration with ORNL, we are attempting to detect certain fission product gases such as xenon and krypton. The relative ratio of these gases can be a sensitive indicator of the TRU average mass that was initially fissioned. While existing accelerator (and nonaccelerator) based assay techniques can determine total "heavy" content, and total fissile content, purely nuclear detection methods have problems as to specific isotope determinations. The RIS method may add another detection scheme that could be applied simultaneously with existing nuclear methods. Some types of wastes may require such additional information for a complete analysis of transuranic content.

The program calls for producing two-photon excitation of, for example, xenon atoms from TRU fission products in the presence of other species. Ensuing photons will produce ionizations, leaving both positive ions and electrons to be detected by charged particle detectors. This year the experimental cell was completed and tested, further calculations of potential background from natural background xenon and from accidental multiple photon off-resonance ionizations were performed, and final selections of tuning crystals were made. The laser system for the program arrived at ORNL the second week in September.

First system runs with xenon and a carrier gas will take place early in CY 1980. LASL is pursuing this TRU monitoring method with a modest effort. Whereas matrix effects will preclude its use for some waste types, it should be applicable to others. Its very high sensitivity for TRU is the reason for these preliminary studies.

PUBLICATION

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F. Slagging Pyrolysis Incinerator k-Effective¹ Calculations and Monitoring Studies

[J. T. Caldwell, T. E. Booth, D. A. Close]

LASL is working on a study in support of the Idaho National Engineering Laboratory (INEL) Slagging Pyrolysis Incinerator (SPI) project. LASL has set up a neutron Monte Carlo program for the primary gasifier, hearth, slag tap, and other regions. The program is running with what amounts to the exact three-dimensional geometry of the system. LASL has calculated (for a 4'-diam x 6'-long isolated slag tap) a multiplication of $M = 1.56$ ($k = 0.36$). This assumed a uniformly distributed plutonium mass of weight density 0.52%. First interpretation is that the slag tap does not allow k-effective to approach unity if one can count on uniform plutonium dispersal. The basic incinerator geometry has also been set up in a single problem and several typical plutonium distributions have been produced with this setup. These are now in the process of evaluation.

LASL also calculated the Feynman variance k-effective monitoring system (passive neutron) based on assumed detector efficiencies of 0.01 to 0.03 (reasonable for the slagger geometry). Things look very encouraging at this point. The basic Feynman variance data are proportional to M^2 for a given detector efficiency. LASL also developed and calculated a system response for a passive Rossi alpha¹ measurement that can be done simultaneously and independently of the passive Feynman variance² measurement. Both methods use the same input data; only the processing is different. The passive Rossi alpha data are essentially linear with M .

Neutron detector optimization studies have begun, both in the direction of optimal detection efficiency per ³He proportional counter and in the direction of minimizing detector dieaway time. (This is required for better Feynman variance or Rossi alpha measurements.) Also performed were a series of passive Feynman variance measurements on multiplying systems of plutonium material ($M \leq 2$). These measurements have confirmed the theoretically expected variation ($\sim M^2$) with system multiplication and indicate the feasibility of using this approach for the basic monitoring of k-effective. LASL also performed pulsed neutron (D + T source) system measurements and has purchased fast-recovery electronics appropriate for use in this system. LASL is evaluating both induced prompt neutron measurements (using the differential dieaway approach — detector $\tau_{1,2}$ much less than characteristic system $\tau_{1,2}$) and delayed-neutron activation for their applicability.

Reference material has been accumulated for use as background for the Feynman and Rossi alpha techniques.

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G. Assay Instrumentation Support for an Electropolishing Facility at Pacific Northwest Laboratory

[D. A. Close, G. O. Bjarke, D. F. Anderson and C. J. Umbarger]

In support of the electropolishing development program at Pacific Northwest Laboratory (PNL), LASL is supplying technical assistance for their various nondestructive assay requirements for special nuclear materials from a waste management perspective. During this calendar year, LASL has designed, constructed, and installed a liquid effluent assay system (off-line), a 1000-cm² alpha-monitoring system, and a prototype low-density waste package assay system (of the MEGAS type and only for demonstration purposes). LASL has also installed a hyper-pure germanium detector system and assisted with a small phoswich detector system (both for low-energy photon detection). LASL has begun preliminary measurements at PNL on a glovebox holdup monitor using a thick NaI detector located outside the box. This system has a detection limit of < 5 g.

PRESENTATION

D. A. Close and T. E. Booth, "In Situ Nondestructive Plutonium Assay for an Electropolishing Bath," presented at the Winter Meeting of the American Nuclear Society, San Francisco, Calif., November 11-16, 1979.

H. High-Density Waste Assay: Another Approach

[J. T. Caldwell]

LASL is developing a pulsed neutron (D + T neutron generator) assay system for an approximately 55-gal-barrel-sized container. The novel feature of this system is a graphite-lined (several inches thick) assay chamber and a neutronically decoupled moderated ³He proportional counter-detection system.

LASL appropriately named this system the Differential Dieaway system. Within the assay chamber, the initial D + T neutron pulse dies away in about 1.2 ms and within the detector region, the same pulse dies away in usually 65-70 μ s — far more rapidly. Thus, in the assay chamber, plutonium may undergo (n,f) reactions induced by the still-present interrogating pulse, leaving no interrogating neutrons within the neutron-detection region. The induced-fission reactions in the assay chamber may then be detected by fast-neutron transport and, subsequently, there is moderation within the neutron-detection region.

In this system an induced-fission event is detected with a highly efficient thermalized neutron detector and without the requirement of coincidence. Preliminary results show a gain in sensitivity of more than a factor of 100 over (n,f) interrogation followed by delayed-neutron counting. Also, the system backgrounds are much lower because one counts for much shorter time periods than is required in delayed-neutron techniques. Work on this project continues in FY 1980.

I. Waste Assay Workshop, October 23-25, 1979

[C. J. Umbarger]

The LASL waste assay instrumentation development and application workshop was limited to DOE contractors and their representatives. Eleven contractors were represented, along with two DOE project offices, Office of Nuclear Waste Isolation (ONWI) and Waste Isolation Pilot Plant

(WIPP). Twenty-one non-LASL participants attended. A summary of the meeting was completed in November and distributed to the attendees. The workshop provided an excellent forum for technology transfer between the assay instrumentation developers and the operational waste managers. Cooperation and collaboration between the many contractors have been greatly strengthened through the meeting. Appendix A contains a copy of the agenda and a list of workshop participants.

J. Alternative Transuranic Waste Management Methods

[L. J. Walker and W. R. Hansen]

The major objective of the evaluation of transuranic (TRU) contaminated waste sites program has shifted to a study of alternative waste management methods. The evaluation and documentation of the methods that are possible will lead to the preparation of an Environmental Impact Statement (EIS) discussing the long-term management options for TRU and low-level waste at LASL. In addition, the development of more general long-term evaluation methods for past, present, or future waste sites is a continuing effort.

Some of the conditions at LASL that have bearing upon the possible alternatives for radioactive waste management are considerably different than at many of the other DOE waste burial areas. For example,

- Low rates of rainfall and snowfall,
- Higher evaporation than precipitation,
- Specific geological and hydrological conditions, and
- Multiple sites within the LASL-DOE controlled areas.

There are six waste areas (active and inactive) at LASL, some of which may contain TRU wastes. The waste areas are all designated by a letter (Fig. 1). The currently active waste area is

Fig. 1.
Possible TRU waste disposal areas.

designated G. The facilities for storage of TRU wastes are located at Areas G and T. The identified alternatives for long-term management of these areas are scheduled for completion in October of 1980.

For a more detailed description of the wastes, the waste areas, and the environmental characteristics at LASL, the interested reader is referred to the LASL Final Environmental Impact Statement, DOE/EIS-0018, and the annual LASL Environmental Surveillance reports.

Other DOE laboratories are also preparing alternatives studies for TRU wastes. There is an effort to standardize the methods and comparisons used by these laboratories and to use the best efforts of each group. However, many of the environmental conditions, waste volumes, and waste constituents may be highly specific to LASL. Therefore, some alternatives may be specific to LASL and differ from those considered by other DOE facilities.

PRESENTATION

W. R. Hansen, A. J. Ahlquist, R. Garde, and J. Umbarger, "Some D&D Experience at Los Alamos Scientific Laboratory," DOE Decommissioning Criteria, Sun Valley, Idaho, September 19-20, 1979.

K. Engineering Studies for TRU Waste Management Alternatives

[G. Maestas and D. C. Nelson]

Engineered waste management alternatives and the costs and engineering associated with their implementation are being defined for existing waste sites at LASL. Efforts have consisted of a preliminary records review to determine volumes and locations of TRU waste at LASL and a selection of alternatives. Inputs to the final alternatives are being drafted.

The object of the review is to determine the locations, volumes, and characteristics of waste disposal areas at LASL containing or potentially containing TRU waste. Documents from other Department of Energy laboratories involved in similar studies were reviewed and personnel at those laboratories were consulted in an effort to evaluate existing methods.

Waste management alternatives selected for the study include (1) continuation of current practices, (2) enhancing containment through engineered improvements, and (3) exhumation of the buried waste and retrieval of stored TRU waste. The continued current practices and enhanced containment alternatives were to apply initially only to buried waste, but now also include stored TRU waste.

The proposed engineered improvements include using a compacted tuff layer of at least 5 ft over waste, river rock riprap over the 5 ft of compacted tuff or a combination of compacted tuff, concrete, and riprap. Calculations for each of the proposed improvements show that enhanced waste containment would be obtained against aircraft crashes, meteorites, earthquakes, tornadoes, plant and animal intrusion, and wind and water erosion. It was concluded that the measures would be virtually ineffective against volcanoes and that intrusion by man would be made only more difficult, but could not be prevented.

The exhumation alternative would involve digging into inactive waste pits known to contain or suspected of containing TRU (>10 nCi/g) waste. The non-TRU waste would be separated and placed back in the pit, and the TRU waste would be transported to a waste processing facility (WPF). The operation would be extremely difficult because of wide variations in waste characteristics and because of the large volumes that must be handled and assayed to locate TRU quantities. Exhumation would have to be carried out at two or more locations simultaneously if it were

to be completed in the 10-yr period originally allocated. Further study may indicate that more time is needed. Retrieval of the stored TRU waste should be considerably easier because the waste is in containers, primarily 55-gal drums and 4- by 4- by 7-ft fiberglass-reinforced polyester-coated wooden boxes.

Another major aspect of the study is the conceptual design of a WPF for treatment of stored and buried waste. The facility now proposed would include an incinerator for combustibles, decontamination equipment for metals, and concrete fixation equipment for ashes and non-metals, noncombustibles. Although the volume of waste retrieved and exhumed could be large (as much as an estimated 13 Mft³), the amounts of actual TRU waste to be processed through the WPF could be as little as an estimated 1 Mft³. Only an estimated 80,000 ft³ are combustible, and ~275,000 ft³ are metals for possible decontamination. The remaining waste could require fixation or compaction and packaging. At present, it is uncertain how much of these wastes might be in concrete or steel drums, and therefore how extensive the required processing might be. Also unclear is whether certain areas (especially Areas A, B, and C) and certain pits might contain TRU-contaminated wastes as opposed to actual TRU wastes. Wastes less than the 10-nCi/gram limit would be placed back into the waste pit and would not require processing or repackaging. Thus, the actual volumes of waste would vary considerably. It is possible that the overall risks and costs could be considerably lower if treatment at the WPF is minimal, if much of the waste is simply repackaged, or if it is placed back in the pits.

Disposal options for the packaged or processed TRU waste include deep-pit disposal at LASL and off-site disposal at a Federal repository.

L. Risk Estimates for TRU Waste Management Alternatives

[J. M. Graf, F. A. Guevara, and W. J. Wenzel]

The radiological risk associated with alternative waste management strategies is being evaluated. The effort thus far has consisted of

- (1) compiling occupational external radiation exposures from an analysis of a 1974-1978 LASL film and thermoluminescent dosimeter (TLD) data of employees involved in radioactive waste disposal-related activities,
- (2) measuring individual external doses incurred in the performance of specific radioactive waste management tasks that are similar to the tasks anticipated in the waste management alternatives, and
- (3) calculating population inhalation doses based on the population density within a 60-mi radius of LASL, site-specific atmospheric dispersion data, and a reference release of 1 Ci/min.

To estimate the occupational dose commitments for the proposed alternatives, LASL monthly film and TLD badge doses for 1974 through 1978 were analyzed for employees working in operations involving TRU materials. Average doses in rems were calculated for LASL and Zia (LASL subcontractor) personnel. These data will be used as guidance for estimating external occupational doses for each alternative. Table I presents the data selected as representative of values for the continue-current-practices alternative. For this alternative, it is assumed that buried and retrievably stored TRU wastes are in a fenced area covered by approximately 3 ft of

TABLE I
AVERAGE REM PER YEAR PER INDIVIDUAL
OCCUPATIONAL EXPOSURES SELECTED
FROM LASL DOSIMETRY RECORDS^a

Year	Operating Unit		
	H-7 ^b	H-8 ^c	Zia ^d
1974	0.145	0.187 ^e	0.116
1975	0.112	0.045	0.187
1976	0.070	0.044	0.102
1977	0.103	0.031	0.088
1978	<u>0.080</u>	<u>0.083</u>	<u>0.121</u>
Average	0.102	0.078	0.123

^aThese are taken from group averages and they include total occupational exposures, not just exposure that is due to TRU materials.

^bH-7 is the LASL Radioactive Waste Management group.

^cH-8 is the LASL Environmental Surveillance group.

^dZia is a contractor that provides LASL craft support.

^eExtreme value, not caused by TRU materials.

layered tuff that has been graded and seeded. To estimate occupational doses for the alternatives, it is assumed that three crews maintain the area, take and analyze samples, and respond to emergencies in the active waste areas. LASL current practices that parallel those activities are carried out by the Waste Management group, the Environmental Surveillance group, and the Zia Company support personnel. Therefore, LASL dosimetry data shown in Table I are used to establish external doses for the continue-current-practices alternative. The composite value, 0.1 rem/man/yr provides a conservative *maximum* value for the alternative because the film and TLD badge data cover all the external exposures the workers encounter in their daily routines, including exposures to sources other than those from TRU radioactive waste management.

Occupational external doses based on TLD measurements of workers engaged in specific radioactive waste handling operations are being studied. The Harshaw High Sensitivity TLD-100, a lithium fluoride chip that provides approximately a factor of 10 better sensitivity than the TLDs used regularly for LASL personnel badge dosimetry was acquired for the study. The High Sensitivity TLD-100 chips are used successfully for LASL environmental surveillance. The preliminary measurements of occupational exposures included workers that were either packaging, transporting, or unloading radioactive waste in routine LASL waste management operations. One study measured the doses of the workers loading retrievable TRU waste into fiberglass-covered plywood boxes, the riggers moving the boxes, and the workers closing and sealing the box tops. In a second study, measurements were made of workers handling hot-cell waste. The activities of the hot-cell operators were monitored while they packaged the waste, transferred the package into a cask, removed the cask from the hot cell, transported the cask to the disposal site, and unloaded the package from the cask by lowering it into a storage shaft. In a

third study, measurements were made during retrievable-waste barrel-stacking operations at area G, the storage-burial site. Each worker was badged with a packet containing four TLD chips. The duration of each individual's task was recorded, the radiation field was measured with a portable instrument, and the TLD doses were correlated with the inventory of TRU handled and the residence time in the field. Preliminary measurements are encouraging but inconclusive because accuracy is difficult to attain for this type of measurement. Future measurements will be improved by using a recently developed Victoreen calcium fluoride chip with the capability to read 0.1 mrem to $\pm 5\%$ accuracy. Occupational external doses values established from direct measurements during specific tasks will provide a valuable reference in assessing radiological risks for the various alternatives and in comparing the doses estimated for each alternative.

The population inhalation reference doses calculations are based on the population density within an 80-km radius of the three major waste burial sites at LASL; TA-21, TA-50, and TA-54. The principal elements for the calculations consist of the computer codes PATHFINDER² and DACRIN,¹ and dispersion coefficient, χ/Q , isopleths. Figure 1 illustrates the methodology for the calculations. The PATHFINDER code provides the population distribution for the surrounding area. DACRIN is used to calculate inhalation doses for the liver, bone, lung, and the lower large intestine as well as for the whole body. The dispersion coefficient isopleths provided by LASL meteorologists are used to calculate the radioactive material concentrations based on the assumed reference release of 1 Ci/min. The calculated reference doses values are the basis for the population doses to be estimated for each alternative and will be appropriately scaled once the scenarios and quantities released have been developed. The calculations include doses from the nuclides ²³⁸Pu, ²³⁹Pu, ²⁴¹Am, ⁹⁰Sr, and ¹³⁷Cs.

Fig. 1.
Population inhalation reference dose calculation methodology.

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M. Development of Evaluation Methods for Transuranium-Contaminated Waste Sites

[M. L. Wheeler and J. C. Rodgers]

A major goal of this program is to evaluate the potential for transporting radionuclides from buried TRU or low-level waste to the biosphere. A key element in this evaluation is the use of detailed computer models that use site-specific data to simulate dominant release and transport mechanisms. A preliminary evaluation¹ has identified intrusion by plants as a possible release mechanism (independent of human involvement). Subsequent redistribution of plant material in soil and litter, followed by surface erosion and stream transport, forms an important potential pathway in the environment.

BIOTRAN², a simulation model developed at LASL, is being used to evaluate the first part of this release pathway. The portion of the model predicting biomass density of typical northern New Mexico forests is undergoing a validation test using data developed under a New Mexico Highlands University contract. Field data of piñon-juniper and ponderosa pine areas are compared with values simulated from site-specific data on climate, soil types, and elevation. In general, there is agreement between observed and simulated values. This validation testing is being extended to include Douglas fir, white fir, aspen, and oak shrub areas.^{3,4}

An independent review and evaluation of BIOTRAN was completed by Colorado State University.⁵ This included an assessment of the biological correctness of the model, a check of the coding, and an overall evaluation. Also, a literature search developed data to support extension of the radionuclide-transport-simulation capability to include uranium and fission products.^{6,7} Revisions and corrections identified by this review are being incorporated in the final documentation of BIOTRAN.²

A joint study was initiated this year with PNL to determine the feasibility of coupling LASL biotic models with PNL models of soil erosion and sediment transport.^{8,9} A limited scale test of these models was performed using data from one of the LASL waste disposal sites located on the rim of a major canyon.¹⁰ The study involved simulation of conditions 100 yr after release of the site to natural revegetation, including vegetative cover and radionuclide uptake, surface runoff and sediment loadings, and finally, instream discharges and radionuclide transport by sediments. There was some difficulty in applying the PNL models to the LASL environs, but in general, the results are encouraging. Meteorological¹¹ and hydrological data developed in environmental surveillance activities over the past several years appear to be adequate to calibrate the simulation models. Furthermore, it appears feasible to couple the various models to describe a sequence of processes from plant uptake to sediment transport of radionuclides. Additional capabilities, such as radionuclide transport in the partially saturated zone and radionuclide dispersion in the atmosphere may be added in the future.

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N. Environmental Surveillance of TRU Waste Management Areas

[D. L. Mayfield, L. J. Walker, and W. R. Hansen]

Retired waste management areas at LASL are being monitored as part of the overall environmental surveillance program. However, a monitoring plan for each waste area to evaluate possible close-in migration mechanisms has been added.

The monitoring plan will

- document migration of radioactive contamination beyond the controlled-access fences of each radioactive waste area.
- assure that all applicable federal, state, and other regulatory criteria are being met.
- document possible changes over extended time periods.

- assure that radioactive waste areas at LASL are being maintained in an environmentally acceptable manner.
- provide data for various environmental assessments prepared by LASL, such as the Annual Environmental Surveillance report and any environmental impact statements required before making changes or additions to activities at LASL.

The general environmental surveillance plan consists of three phases: a preliminary phase, a detailed phase, and a routine surveillance phase. An individualized plan will be written for each specific waste area. Each phase will consist of several steps.

The first step will be a cursory search of the LASL data base for all environmental monitoring data applicable to the waste area under study. The second step will be a surface monitoring of the waste area and the immediate area surrounding the controlled-access (fenced) area, using portable radiation detection instruments. The third step will consist of an engineering survey of the area with a randomized point of origin to obtain a randomized grid. The fourth step will be obtaining surface soil and vegetation samples at selected points on the sampling survey grid. Surface vegetation samples will be obtained which represent three different root depths, samples of annual plants, and samples of perennial plant species in their second year of growth. The fifth step will be to obtain subsurface soil samples at depths of 1 to 10 cm.

Subsurface soil samples will be obtained using split-spoon sampling devices. The split spoon is a tool that is driven to a predetermined depth where the spoon collects a sample. The tool is withdrawn from the hole to retrieve the sample. The sample depths will be based on representative depths of original trench burials and will be deep enough below the original trench depths to adequately sample possible plume dispersions below and to the sides of the original trenches. All holes will be filled or capped to prevent transport downward, such as water movement, or will be cased and/or capped for future use for moisture monitoring, *in situ* radiation measurements, or well-hole logging purposes. In no case will cores or holes penetrate into the actual waste materials. The preliminary phase will consist of the minimum number of samples needed to obtain statistically valid results. If the results of this work indicate that additional data are needed, the second phase will be initiated.

The second or detailed phase will consist of more thorough and detailed monitoring and sampling, with considerably more samples and much closer sampling intervals along the sampling grid.

The third phase will consist of the same steps and will be conducted in the same manner as the preliminary phase but will be carried out at periodic, scheduled time intervals. Typically, each of the waste areas would be studied once every year over the entire period of LASL institutional control. In this manner, a considerable data base would be accumulated on each waste area, allowing the early identification of possible long-term trends and the prompt recommendation of any required remedial actions.

Based on data from a 1977 survey of waste area B and current analysis of the data from the sample library, the radioisotopes of concern in the surveillance plan include: ^{238}Pu , ^{239}Pu , ^{240}Pu , and ^{241}Pu ; ^{241}Am ; ^{232}Th ; ^{233}U , ^{235}U , ^{238}U , and total uranium; ^{227}Ac ; ^{226}Ra ; ^{137}Cs ; ^{90}Sr ; and ^3H . Data from this 1977 study indicate the possible surface distribution for radioactivity as shown in Fig. 1. The first or preliminary phase has already been started for waste disposal Area B, and will be continued through CY 1980.

Fig. 1.
Surface distribution for radioactivity (in pCi/g) in eastern portion of waste disposal area B.

O. Controlled-Air Incineration

[W. E. Draper, R. A. Koenig, J. M. Newmyer, A. S. Neuls, L. A. Stretz, and C. L. Warner]

1. Project Perspective. In CY 1975, a dual-chamber incinerator was purchased to develop and demonstrate controlled-air incineration (CAI) technology coupled with a high-energy aqueous off-gas treatment system. Major considerations in the selection of this technology for development included availability of commercially available components, flexibility in handling various waste feed compositions, ease of combustion-rate control, completeness of combustion, and low particulate emissions.

Assembly and equipment modifications for application to TRU solid-waste products occurred during CYs 1976 and 1977. In CYs 1978 and 1979, Phase I, the nonradioactive process operation was successfully completed and Phase II, the radioactive waste treatment operation was begun. The first Phase II experimental run was completed in December 1979. The immediate goal of the CAI project is to develop and demonstrate an effective, safe, and reliable process for volume reduction and chemical stabilization of TRU solid wastes, and to provide the specifications and operating procedures for appurtenant equipment.

The CAI process involves the subsystems (1) feed preparation and introduction, (2) incineration, (3) off-gas cleanup (4) scrub solution recycle, and (5) ash removal.

Sealed 0.06-m³ cardboard boxes of combustible waste products, packaged at the waste source and transported to the Treatment Development Facility (TDF), are introduced to the process with a receiving glovebox. The boxes are assayed for TRU content and x-ray scanned for noncombustible materials. If necessary, the packages are opened for removal of the noncombustible items. The cartons are then resealed and moved to a storage glovebox until a sufficient number has accumulated to support several hours of incineration. Two ram feeders, operating in series, transfer the packages from storage to the lower chamber of the incinerator.

A natural-gas burner ignites the waste as it is charged, and air (slightly greater than stoichiometric requirements) is introduced, sustaining combustion. Excess air and supplemental heat are introduced to complete combustion in the upper chamber. Residual ash is removed from the lower chamber by gravity or vacuum ash-removal systems.

The high-energy aqueous off-gas scrub system removes the particulates and acid gases in the upper chamber exhaust, which may include radioactive contaminants. Scrub solution recycle minimizes liquid blowdown (waste to final treatment) from the off-gas subsystem.

2. Phase I Completion (Nonradioactive Testing). Six simulated (cold) waste experimental test runs of the CAI process were completed during Phase I, including two in 1979. During this period, the majority of technical and equipment concerns related to TRU waste incineration were addressed. All construction, assembly, and modifications required for radioactive operations were completed. Tests during Phase II, contaminated-waste processing, are intended to determine the distribution of radionuclides within the process and the effectiveness of containment provisions.

Design, construction, and installation tasks completed this year included a gravity ash-removal system to permit continuous incinerator operation and zoned ventilation partitions for contamination control in the process area. A major modification to the CAI test program was initiated with the preparation and installation of refractory samples in the floors of both combustion chambers and in the walls of the lower chamber (Sec.0.4).

Experimental Run No. 5, a 140-h system test, was completed on July 25, 1979. Objectives of the run were

- equipment/instrumentation checkout of
 - a. the new main burners,
 - b. burner controls,
 - c. gravity ash-dropout system,
 - d. the secondary coolant loop,
 - e. the data acquisition system, and
 - f. the switchover circuitry for the induced-draft blowers;
- refractory sample cure;
- elimination of quench sump air bleed;
- operation on design basis feed and ash accumulation;
- acquisition of base-line operating data; and
- determination of effect of ash dropout frequency on ash-carbon content.

Run No. 5. was adequately successful in light of the large number of major equipment changes made after the preceding run. Specific accomplishments included satisfactory refractory sample cure, elimination of air bleed into the quench sump, effective gravity ash discharge, and effective operation of the secondary coolant loop, data acquisition equipment, and process fan switchover system. Much improved flame characteristics were realized with the new premix burners, and the fast shutdown control system performed satisfactorily. Base-line operating data, however,

were compromised significantly by malfunction of the ultraviolet (uv) flame sensor in the lower chamber and inadequate calibration of the gas and air-flow measuring elements. Location of the uv sensor was changed since the previous runs to accommodate LASL Safety Group recommendations. Inadequate ventilation in the new configuration caused sensor overheating and burner shutdowns. Flow sensors could not be calibrated because of the late delivery of needed equipment.

A total of 1200 lb of synthetic design basis feed (35% cellulosic, 23% polyethylene, 12% polyvinyl chloride (PVC), and 30% rubber) was burned during the run. Charging operations were interrupted by the flame sensor problem, but a 100-lb/h feed rate was maintained for about 4 h. The ash, considerably different from previous runs, was very dark and contained numerous agglomerations. Shorter residence times as a result of the gravity ash system testing and inadequate underfire air caused by erroneous flow sensor signals were responsible for the higher carbon content.

Liquid samples from the scrub system showed extremely effective removal of acid gases. Chloride and sulfate peak concentrations of 1210 ppm and 122 ppm, respectively, in the absorber sump corresponded to condensate maximums of 5 ppm Cl⁻ and 0.15 ppm SO₄²⁻.

As observed in earlier runs, scrub system particulate removal efficiency was completely satisfactory. The process HEPA filters showed a pressure drop of less than 1.0 in water column at the end of the run despite process upsets caused by numerous burner relights. The filters were inspected and reinstalled for use in Run No. 6.

Postrun inspection revealed no equipment deterioration or failure. A number of minor deficiencies such as piping leaks, malfunctioning sensors, and improper control settings were corrected. A delumper was added to the gravity ash hopper to assure that bridging problems would not recur. Flow element calibrations were made on receipt of the needed instrumentation.

Experimental Run No. 6, the final Phase I CAL experiment, was an 85-h run completed during the week of September 28, 1979. Specific objectives included

- equipment/instrumentation checkout of
 - a. uv sensors,
 - b. gravity ash-dropout system,
 - c. secondary coolant loop,
 - d. process liquid pH control system, and
 - e. gas flow elements;
- operation of design basis feed and ash accumulation;
- definition of base-line operating conditions at 100 lb/h feed rate, and
- study of the effect of ash-dropout frequency on carbon content of ash.

Overall, the operation was most satisfactory. A total of 1600 lb of design basis feed was burned, the most of any run so far. The ash contained excessive carbon indicating a need for much slower ash-drop frequencies. Some minor problems were experienced, but none that would seriously affect planned contaminated waste operations.

The gravity ash-receiving and transport systems worked well even with friable material. Properly calibrated air and fuel gas flow elements resulted in far smoother process control. Other positive results included process operation using design basis waste at feed rates of 60, 80, and 100 lb/h, good performance of the secondary coolant loop and successful recording of run data by the acquisition system. Maximum chloride concentrations in the absorber and condensate sump were 2800 and 20 ppm, respectively, indicating excellent acid gas removal. Most significantly, the process HEPA filters service time was extended to a total of 225 h.

3. Phase II Experimental Run (CAI Transuranic Waste Demonstration). Goals to be completed during this phase are (1) demonstration of the CAI process with TRU wastes and (2) documentation of the process including equipment specifications, operating procedures, and test results. Initial radioactive testing, completed during December 1979, used low-level wastes (< 10 nCi/g) to minimize the consequences of potential problems. The demonstration run, expected in March 1980, will use TRU wastes (> 10 nCi/g).

In addition to process development needs, a number of safety requirements were completed during 1979, leading to Phase II. These included approval and issuance of the Final Safety Analysis Report and the CAI Operating Manual, including applicable standard operating procedures. Personnel were trained in the use of anticontamination clothing, health physics instrumentation, and radiation safety procedures.

The principal objective of CAI Run No. 7 was to test contaminated waste in the process. Randomly selected low-level waste from the LASL Plutonium Facility was used. The feed preparation line was operated without problems for one day before starting the CAI. Boxes were assayed, x-rayed, sorted and repackaged. Considerable sorting experience was gained because the LASL waste is currently segregated into compactible and noncompactible fractions rather than combustible/noncombustible. About 34 lb of noncombustibles were removed. A total of 130 ft³ of waste weighing approximately 470 lb was incinerated over a 13-h charging period. The charging rate was intentionally limited to minimize potential problems while experience was gained with contaminated materials. Operations were the smoothest to date and no contamination problems were observed. Approximately 1 ft³ of ash weighing 16 lb was removed and packaged after the run. The resulting weight and volume reduction factors of 29 and 130, respectively, were high because of the low-density and high plastics content of the waste. Secondary solid wastes, such as liquid cartridge filters and surgeons gloves, generated by the run were also incinerated.

4. Combustion Environment Refractory Studies. A refractory evaluation project was begun this year to provide a detailed performance assessment of selected incinerator lining materials for the CAI to check physical degradation, chemical attack, and plutonium uptake. With the assistance of C. Semler of the Refractories Research Center, Ohio State University, six materials were selected (see Table I), two of which are being tested both with and without stainless steel fibers imbedded in the material. In addition, two coating materials were selected for testing and applied to the refractory in the lower chamber hearth area.

Existing refractory was removed from the entire hearth of the lower chamber and a 2- by 2-ft section of the upper chamber floor. Relining work in both areas began with installation of 2-in.-thick insulating board on which a flat floor of Harcast ES castable refractory was poured. Thermocouples were embedded to allow temperature measurements at sample surfaces, between samples and the castable refractory, and between the castable floor and the insulation. Four-inch core samples were also installed on the lower chamber walls to determine refractory performance in vapor space.

Refractory samples (9- by 4 1/2- by 5-in.) were made using special molds obtained from the LASL Ceramics Section. Plastic refractories were rammed into the molds using an air hammer with a special 2- by 2-in. flat ram. Castable materials were cast into the molds on a vibrating table to assure proper filling. For samples in which stainless steel fibers were incorporated, the fiber and material were weighed to produce a fiber loading of 4 lb/ft³ based on mixed refractory volume. All bricks were pre-fired to achieve dimensional stability in the refractory and to avoid damage to the sample installation. Samples in the lower chamber walls were formed directly in the core holes in the lower chamber wall.

For purposes of sample layout, the lower chamber hearth was subdivided into two regions — the combustion zone near the feed end of the incinerator and an ash zone near the discharge end.

TABLE I
REFRACTORY SAMPLE MATERIALS AND COATINGS

<u>Sample No.</u>	<u>Refractory</u>	<u>Type</u>	<u>Manufacturer</u>	<u>Approx Composition (Major Constituents)</u>
1	Super Hybond	Plastic	A. P. Green	45% Al ₂ O ₃ , 50% SiO ₂ , 1.5% Fe ₂ O ₃ , 2% TiO ₂
2	Jade Pak 88-P	Phosphate-bonded plastic	A. P. Green	87% Al ₂ O ₃ , 2.5% SiO ₂ , 10% Cr ₂ O ₃
3	High Alumina Plastic	Plastic	A. P. Green	73% Al ₂ O ₃ , 22% SiO ₂ , 2.5% TiO ₂ , 1.5% Fe ₂ O ₃
4	Harcast ES	Dense fire-clay castable	Harbison Walker	55% Al ₂ O ₃ , 36% SiO ₂ , 3% CaO, 2% TiO ₂
5	Harcast ES/SS Fiber	Dense fire-clay castable	Harbison Walker	Same + 302 SS fiber ^a
6	Taycor 414 Hydrocast	Hydraulic setting castable	Didier-Taylor	96.5% Al ₂ O ₃ , 2.7% CaO
7	Taycor 414/SS Fiber	Hydraulic setting castable	Didier-Taylor	Same + 310 SS fiber ^a
8	Tiger 911	Chemical bond mullite plastic	Didier-Taylor	68% Al ₂ O ₃ , 26% SiO ₂ , 1.7% TiO ₂ , 2.5% P ₂ O ₅

<u>Coating No.</u>	<u>Material</u>	<u>Manufacturer</u>	<u>Used on Sample No.</u>
1	Mulset BF	Babcock & Wilcox	1,3,4,5,6,8
2	TASIL 101	Didier-Taylor	2,7

^aSS fiber from National Standard Co.

1911 Fibers - 302 SS
2520 Fibers - 310 SS.

Samples were randomly placed within the zones; all materials appear at least once with a coating and once without (Fig. 1). During Experimental Run No. 5, the first following installation of the refractory samples, chamber temperatures were increased slowly to cure the emplaced samples.

5. Gravity Ash-Dropout System. In the latter part of 1978, design efforts were initiated for the fabrication and installation of a gravity ash-dropout system (GADOS) that would permit continuous operation of the CAI. Although the existing vacuum ash-removal system (VARS) functions effectively, its operation has the inherent limitation of requiring complete incinerator cool-down before residue is removed. The GADOS can be actuated with the expected range of lower chamber temperatures to effect ash removal with no disruption of the incineration process. Intended to complement each other, the vacuum and gravity systems share common transport and packaging equipment.

Fabrication and installation of the GADOS began in early 1979 and were essentially completed in May. When triggered by an operator, a refractory-lined door in the lower chamber floor opens and ash falls into a stainless steel receiving hopper (Fig. 2). The cooled ash is transported pneumatically to the packaging station (Fig. 3).

Initial operational testing occurred during Run No. 5 and proved successful except for solids bridging problems in the receiving hopper. To correct this deficiency, a delumper (Fig. 4) was added and the complete system operated smoothly during Run No. 6. The proven GADOS will serve as the primary ash-removal mechanism with the VARS being used for thorough cleanout of both chambers for inventory control and inspection purposes.

Fig. 1.
Refractory sample distribution.

Fig. 2.
Refractory-lined door.

Fig. 3.
Packaging station.

Fig. 4.
Delumper.

PUBLICATIONS AND PRESENTATION

C. L. Warner, Ed., "Final Safety Analysis Report for the Transuranic Contaminated Solid Waste Treatment Development Facility," Los Alamos Scientific Laboratory report LA-7971-MS (July 1979).

W. C. Whitty, "An Evaluation Procedure for Radioactive Waste Treatment Processes," Los Alamos Scientific Laboratory report LA-8052-MS (November 1979).

L. C. Borduin, "U.S. High Temperature Incineration R&D for Treatment of Alpha-Bearing Wastes," presented to NEA/OECD Workshop on High Temperature Incineration of Alpha-Bearing Wastes, Centre d'Etudes de l'Energie Nucleaire, Mol, Belgium, November 1979.

P. Ash Immobilization

[L. A. Stretz]

The 2-yr study of ash immobilization methods was aimed at identifying and evaluating alternative final residues forms compatible with acceptance criteria for permanent radioactive waste repositories. Because this project was not funded for FY 1980, testing was terminated at the end of FY 1979. Work completed at that time is being summarized in an informal report.

The initial phase of this work involved a thorough literature survey of existing fixation techniques and extensive interaction with other DOE site personnel involved in this study area. In June 1978, a meeting held at LASL established the high-priority need for a standard leachability test and generated a draft list of desired physical and chemical properties for satisfactory waste forms.

At LASL, study was oriented along two paths. First, quantities of CAI ash from Phase I experimental runs were forwarded to Battelle PNL for incorporation in a glass matrix and to Mound Laboratory to obtain pressed concrete/ash samples. It was intended that additional samples in glass, concrete, ceramic, and other potential forms would be solicited from other sources as ash products became available through subsequent Phase I and II runs.

Second, a process using clay as the fixation medium was selected for bench-scale evaluation. A statistical, experimental design was completed for establishing the role of eight variables—clay type, ash/clay ratio, water content, pelletizing pressure, pressure duration, degree of dehydration, sintering temperature, and duration of sintering — on product performance parameters including crush strength, impact resistance to breakage, abrasion resistance, and leachability. Testing procedures included microstructure characterization through photomicrography, impact testing, and leaching. Data obtained from this test array are shown in Table I. Significant effects on crush strength were observed to be a result of clay type, ash/clay ratio, and sintering temperature. Abrasion resistance was apparently influenced by ash/clay ratio and sintering temperature; impact resistance showed marginal influence as a result of the clay type or significant influence from the ash/clay ratio. Effects from the other variables in the range tested were either not statistically significant or were masked by the scatter in the data.

Preliminary volume reduction data from the initial clay pelletizing process indicated compaction ratios approximating 4.5:1 for ash resulting from a CAI mixed-waste experiment. The primary volume reduction (that is, not including secondary wastes) realized by incineration of a waste package to vacuum ash discharge was 40:1. Therefore, the overall CAI-clay process volume reduction approximates 180:1. It must be emphasized, however, that these data are preliminary and additional work is required to verify factor dependency on waste composition and immobilization conditions.

TABLE I
CLAY/ASH PELLET SCREENING TEST SUMMARY

Variables Tested	Effect On:		
Variable (high/low)	Crush Strength	Abrasion Resistance (% Weight Loss)	Impact Resistance (1 kg dropped 1 m; % passing 40 mesh)
Clay type (tb/bL) ^a	-660 psi	NS ^b	+0.7%
Ash/clay ratio (9:1/2.3:1)	-1140 psi	+0.25	+2.67%
Water content (7 g/3 g)	NS	NS	NS
Forming pressure (10 k lb/5 k lb)	NS	NS	NS
Pressing time (5 min/1 min)	NS	NS	NS
Dehydration (yes/no)	NS	NS	NS
Sintering temperature (1200°C/1000°C)	-1665 psi	-0.68	NS
Sintering time (3 h/1h)	NS	NS	NS
Range in crush strength:	highest 4600 psi ^c	lowest 100 psi	
Abrasion resistance:	best 0.01-g loss (0.04%)	worst 0.36-g loss (1.4%)	
Impact resistance:	best 1.44% past 40 mesh	worst 8.9% past 40 mesh	

^atb = tennessee ball; bL = bentolite "L" (Georgia Kaolin Co.).

^bNS = not significant or masked by scatter in data.

^cNew pellet with sifted ash had crush strength of 11 000 psi.

Leach testing on samples that performed well in crush, abrasion, and impact tests were initiated but not completed. Sample cuttings, specimen polishing, and photomicrography of the samples were about 50% completed by the end of September.

Q. Interim Waste Storage Studies

[A. Zerwekh]

There have been experimental studies to obtain data on radiolytic gas generation for application to radioactive waste storage conditions since the mid-1970s. Originally, the experiments were pursued along two lines: monitoring and sampling of radioactive waste emplaced in a LASL interim storage trench, and analyses of simulated waste matrices contained in laboratory experimental cylinders. The latter portion of this study was completed in FY 1978.

The evaluation of TRU waste retrievably stored at LASL as of the end of CY 1977 is now also complete. Data generated by this study include average and maximum container weights, average densities for waste categories, and heat generation rates for the waste inventory.

A total weight was estimated by determining average weights for a container type in each of the 27 categories and extending this measurement to all containers. The total gross weight of the LASL TRU waste inventory is estimated at 1.2×10^6 kg. An average density for the entire inventory is 0.8 g/cm^3 (or about one-third that of natural salt). The total curie content, as well as the heat production rate, is dominated by heat-source plutonium with ^{241}Am and weapons-grade plutonium being the only other significant contributors. The heat production rate (nondecay corrected) for the entire waste inventory is 2.6 kW, which yields an average power of only 1.6 W/m^3 . The TRU waste in storage is contaminated with actinides at a level of $7.8 \times 10^{-2} \text{ Ci/g}$ of waste.

PUBLICATION

A. Zerwekh, "Gas Generation from Radiolytic Attack of TRU-Contaminated Hydrogenous Waste," Los Alamos Scientific Laboratory report LA-7674-MS (June 1979).

R. Controlled-Air Incineration—Commercial Wastes

[A. S. Neuls]

The central objective of this program is to adapt the CAI TRU waste treatment process to nuclear waste volume reduction efforts within the commercial industry. Program goals include incineration of the spectrum of combustible commercial wastes (that is, solid, liquid, and resinous materials), preparation of a conceptual design incorporating remote system provisions, and technology transfer as requested by the several sectors of the industry. Because of funding limitations during FY 1979, emphasis was placed primarily on technology transfer.

Among those requesting CAI process and general waste management input from LASL staff were representatives of the General Electric Company, Westinghouse Corp., Gilbert/Commonwealth, Chem-Nuclear Systems, Stone and Webster, Helix Process Systems, Lockwood-Greene, Associated Technologies, Inc., and Sargent and Lundy. At present, the Nuclear Fuels Division of Westinghouse is constructing a uranium scrap incineration system that incorporates LASL CAI design input. Commercialization of the CAI for low-level waste treatment, discussed separately in following paragraphs, is likewise under consideration.

A member of the LASL Waste Management Group is serving as a member of the American Nuclear Society (ANS) Standards Committee on Volume Reduction of Low-Level Wastes (ANS 40.35). Active involvement with this committee should improve our awareness of commercial industry's waste treatment technology. General discussions during the first meeting of this committee emphasized cost effectiveness and licensing requirements as two hurdles facing the industrial application of incineration as a routine radwaste treatment technique.

In FY 1980 and beyond, program efforts will follow two general paths: demonstration and technology development. Demonstration activities will focus on the goal of providing data and experimental support needed to realize engineering demonstration at a commercial nuclear site(s). Development activities will address technical concerns regarding alternative feed mechanisms, off-gas system performance, and shielding requirements. Following demonstration completion, CAI staff and equipment are viewed as a technical resource available to address specific industrial waste management problems.

The CAI concept, as modified and developed at LASL for combustion treatment of defense program-generated solid wastes, has widely recognized potential for many other waste management applications. A CAI unit is presently being used at Lawrence Livermore Laboratory for disposal of carcinogen-contaminated biological research wastes. A second unit is under construction at the Westinghouse Fuels Fabrication Plant in Columbia, S.C., for uranium recovery and waste volume reduction.

During CY 1979, closer and more stringent regulation of commercially operated low-level radwaste disposal sites caused nuclear facilities to more actively consider alternative means for waste volume reduction. Many of these facilities have concluded that incineration of combustible wastes holds the most significant potential both to reduce shipped quantities and to improve general handling safety. Also, increasing disposal and transportation charges are improving the cost effectiveness of incinerator installations.

Discussions regarding a possible DOE/industry cooperative venture have been held with several commercial firms. A nuclear process equipment vendor is proposing CAI demonstration at a nuclear utility site. Should a contract result from this proposal, nuclear utility/vendor/DOE funding would be used to achieve commercial demonstration for low-level waste treatment.

LASL's role in this venture would consist of providing base technology data and specifications, planning and conducting experiments to provide needed operating data, design review in an advisory capacity, and providing assistance during startup, cold testing, and demonstration. Modifications to permit controlled burning off of liquid wastes and to achieve adequate removal of volatile fission products from the incinerator off-gas will be primary development objectives in support of the demonstration. Completion of this planned demonstration should prove to be a very effective and thorough transfer of DOE-sponsored technology development.

S. Waste Management Requirements for Alternative Fuel Cycles

[T. K. Thompson]

The purpose of this program is to investigate the nonhigh-level waste implications of four alternative fuel cycles currently under investigation in the United States. These fuel cycles are (1) the once-through fuel cycle with no reprocessing and indefinite storage of spent fuel, (2) plutonium recycle in an LWR with spent fuel reprocessing, (3) plutonium recycle in LWRs using a coprocessing method for reprocessing the spent fuel, and (4) thorium/uranium fuel cycles as exemplified by the high temperature gas reactor (HTGR) and the light water breeder reactor (LWBR).

The fuel cycles under investigation were first defined based on a 1000-MWe output from the reactor. Various reactor-supporting components of the fuel cycle (such as uranium mining, milling, refining, and enrichment) were then defined for each cycle. This assured that there was a common basis for evaluating the wastes produced from the various fuel cycles. Figure 1 is a flow sheet for the thorium-based cycle employing an LWBR. Fissile material flows expressed as metric tons uranium, plutonium, or heavy metal that includes both, are shown. The LWBR reactor is typified by the thorium recycle experiment being conducted in the Shippingport reactor. Figure 2 shows corresponding fissile material flows for the HTGR thorium fuel cycle. Figure 3 is the plutonium fuel cycle using an LWR with produced plutonium being recycled as mixed oxide fuel. The reprocessing step in this cycle can either be conventional reprocessing, such as a Purex flowsheet, or can be based on a coprocessing concept where the uranium and the plutonium are never completely separated outside a secure area. This variation of the plutonium fuel cycle was investigated because it has less diversion potential for fissile plutonium than the conventional Purex reprocessing. Figure 4 shows the fissile material flow rates for a typical LWR once-through

Fig. 1.

LWBR prebreeder fuel cycle with fissile material flows (1000-MWe output basis).

fuel cycle with no processing or recycling of uranium and plutonium and indefinite storage of the spent fuel. This cycle is presently the one employed in the United States. The LWR used was a composite reference reactor based on the electrical output being generated by 0.67 pressurized water reactors and 0.33 boiling water reactors.

Waste generation source terms have been formulated for each component of the various fuel cycles. The wastes have been normalized on a per metric ton heavy metal basis for all steps other than enrichment for ease of calculations. The enrichment step used a normalization factor based on a separative work unit (SWU). The final report on this project is in preparation where the results are discussed in detail.

PUBLICATION

Papp Reiner and Hamid Mortazavi, "Characterization of Nonhigh Level Waste in an LWR Fuel Cycle Based on Coprocessing of Spent Fuel—Phase Three," University of Arizona report (September 1979).

Fig. 2.
HTGR fuel cycle with fissile material flows.

Fig. 3.
Coprocessing fuel cycle with fissile material flows.

Fig. 4.
Once-through fuel cycle with fissile material flows.

APPENDIX A

FINAL AGENDA LASL/DOE RADIOACTIVE WASTE ASSAY INSTRUMENTATION WORKSHOP

October 23-25, 1979
Los Alamos, New Mexico

Tuesday, October 23, 1979

8:30 Opening and Welcome
8:40 DOE Program Overview, A. Camacho (DOE)
9:00 ONWI Program Overview, E. Lang (AGNS)
9:20 WIPP, P. O'Brien (SLA)
9:40 LASL, W. Maraman
10:00 Break
10:20 Mound Facility, A. Campbell
10:40 ORNL, N. Cutshall
11:00 RFP, C. Wickland
11:20 INEL, K. McKinley
11:40 Rockwell Hanford, C. Manry
12:00 SRL, R. Hooker
12:20 Lunch
1:15 Panel: Summary of Waste-Handling Programs
2:15 Nuclear Safeguards Measurement Techniques For Waste Management, R. Walton, (LASL)
2:45 Break
3:05 Low Density Waste Assay, D. Close (LASL)
3:30 Crate Counting at RFP, R. Harlan (RFP)
3:55 Passive Counting of Large Packages, J. Caldwell, (LASL)
4:20 General Discussion
5:00 Reception

Wednesday, October 24, 1979

8:30 Opening
8:35 Application of Accelerator Technology to Waste Assay, M. Cates (LASL)
9:10 LASL Accelerator-Based NDA Programs, L. Franks (EG&G Santa Barbara)
9:40 INEL NDA Program, E. Nieschmidt (INEL)
10:10 Break
10:30 Real Time Criticality Monitoring for Waste Incinerators, J. Caldwell (LASL)
11:00 Rockwell Hanford NDA Program, L. Bruns (Rockwell)
11:40 Lunch
1:15 Small Package Counting at PNL, R. Brodzinski (PNI)
1:45 Active/Passive 55-Gallon Drum Counters, T. Crane (LASL)
2:15 Break
2:35 Tour: LASL NDA Instrumentation for Nuclear Safeguards and Waste Management
7:30 Dinner

Thursday, October 25, 1979

9:00 Panel: Waste Assay Matrix, Accuracy, Sensitivity Interference, and Container Size Considerations
10:00 Break
10:20 Panel: Waste Assay Applications Today; Needs and Directions Tomorrow
11:30 Adjournment

Program Chairman: John Umbarger
Health Research Division
FTS: 843-6249
Comm: 505-667-6249

LASL ASSAY INSTRUMENTATION WORKSHOP ATTENDEES

David Anderson, LASL H-1
H. W. Arrowsmith, PNL
George Bjarke, LASL H-1
Jim Bresson, WIPP Project Office
Ron L. Brodzinski, PNL
Les Bruns, Rockwell Hanford
John Caldwell, LASL Q-2
Al Camacho, DOE GTN
Allen R. Campbell, Monsanto
Mike Cates, LASL P-14
Eidon Christensen, LASL CMB-11
Don Close, LASL H-1
Tom Crane, LASL Q-1
Jack Crowell, LASL H-1
N. M. Cutshall, ORNL
Kathi DeGasperi, LASL H-1 (Registrar)
Larry Franks, EG&G Santa Barbara
Conrad Gutierrez, LASL CMB-11

R. A. Harlan, Rockwell Rocky Flats
R. L. Hooker, SRL
Tom Keenan, LASL H-7
J. Kordas, LLL
Ernest J. Lang, Allied Gen'l Nucl. Service
George D. Lehmkuhl, Rockwell Rocky Flats
C. W. Manry, Rockwell Hanford
Kirk McKinley, EG&G Idaho
Lee Neher, LASL P-14
E. B. Nieschmidt, EG&G Idaho
Bruce Noel, LASL WX-10
Paul O'Brien, Sandia Albuquerque
Paul Phelps, LLL
J. L. Pigg, EG&G Santa Barbara
P. D. Randolph, EG&G Idaho
John Umbarger, LASL H-1 (Program Chairman)
Roddy Walton, LASL Q-1
Charles E. Wickland, Rockwell Rocky Flats